Li Wall Fusion - No alternative, No other option

(Flowing Liquid Lithium ($^{7}_{24}$FLiLi) - the key to fusion relevant plasma regimes)

Leonid E. Zakharov[1], Xujing Li[2], Sergei A. Galkin[3], Luca Guazotto[4]

[1] *Princeton University, PPPL, MS-27, PO Box 451, Princeton, NJ 08543-0451*
[2] *Institute of Computational Mathematics and Scientific/Engineering Computing, Academy of Mathematics and Systems Science, Chinese Academy of Sciences, P.O. Box 2719, Beijing 100190, China*
[3] *FAR-TECH, Inc. 10350 Science Center Dr. Bld 14, St 150, San Diego, CA 92121*
[4] *University of Rochester, 500 Joseph C. Wilson Blvd. Rochester, NY 14627*

Theory Dept. Research & Review Seminar

May 30, 2014, PPPL, Princeton NJ

*This work is supported by US DoE contract No. DE-AC02-09-CH11466.

Abstract

In defiance to many "fantastically incorrect statements"[1] of opponents fusion propaganda (which is intended to provide energy from "seawater"[2] while being "unaware of any major project failure in magnetic fusion research"[3]) often uses a trick of making people feel fool in front of "computer simulations of plasma turbulence which helps scientists predict plasma behavior"[3].

In fact, these simulations and the three decade long obsession of FES with the core transport, were critical in termination progress in fusion. During the last 17 years the fusion program followed exactly the path predicted by "The theory of the failure of magnetic fusion"[4] (LZ, 2004), i.e., from progress to stagnation, and then to degradation, when science no longer plays a role.

At this point the result is devastating. After 3-4 decades of development:

- ***(a) confinement theory with its 3-5-D numerical codes has no idea where the confinement zone is in tokamaks,***
- ***(b) the macroscopic stability codes simulate the free boundary plasma as "salt" water, mixed with halo-currents,***
- ***(c) there is not even a basic understanding of the plasma edge and pedestal region,***
- ***(d) the "miraculous" edge transport barrier has created an entire industry of cooking shear flow stabilizations, pedestal bootstrap currents, peeling-ballooning edge stability, screening of RMP, etc.***

The energy "vision" of FES (except its energy from "seawater") is simply ridiculous. After 15 years of existence, FES failed not only in the energy aspects, but even in science. The situation with FES can only get worse.

In contrast, the basic level of science of magnetic fusion has been created in a separate, essentially underground effort of LiWF concept. It provided a much deeper understanding of the tokamak plasma and now raises the necessity of a separate program which would aim toward a P_{DT}=100-200 MW DEMO device with the electric Q factor exceeding unity.

[1] Stewart Prager, Richard Hazeltine, "Rohrabacher's Comments on Fusion Research Are "Misinformed"", APS News, August/September 1995 (Volume 4, Number 8) `http://www.aps.org/publications/apsnews/199508/letters.cfm`

[2] Stewart Prager, "How Seawater Can Power the World", NYTimes, The Opinion Pages, 07.11.2011 `http://www.nytimes.com/2011/07/11/opinion/11Prager.html?_r=0`

[3] Stewart Prager, "The Way Forward with Magnetic Fusion Energy", NYTimes, The Opinion Pages, 11.19.2012 `http://dotearth.blogs.nytimes.com/2012/11/19/in-defense-of-sustained-research-on-fusion/`

[4] Leonid E. Zakharov, "The theory of the failure of magnetic fusion", APS DPP-2007 `http://http://w3.pppl.gov/~zakharov/APS-07F.pdf`

Contents

1 LiWall Fusion (LiWF): the scientific approach to fusion

1. ***1998, Dec. 03 - Bob Woolley informed Theory Dept that*** $\simeq 0.5$ *m of flowing Li (a) screens walls from neutrons, being (b) undestructible, (c) impossible to activate, (d) electro-magnetically controlled, and (d) can bread T.* ***Magnetic propulsion of LiLi was invented.***
2. ***1998, Dec. 5-18 - Basic properties of Li was understood.*** *(a) Li flows, convection, propulsion was understood (thanks to E.Muraviev). (b) Li pool was ruled out as a divertor target. (c) Outstanding* D^+ *pumping by Li coating on T-11M was learned, (d) small Li sputtering by ions was learned (from Illinois)*
3. ***1998, Dec. 23 - 5 min phone conversation with S.Krasheninnikov:***

   (a) flat core temperature regime due to the Li pumping plasma particles was explained,
   (b) the confinement problem seemed to find its appropriate solution.

4. ***1999, Jan.08, Jan 18 - first two talks*** *on new plasma regimes to PPPL and to ITER.*
5. ***2001 new understanding of the*** T_e^{edge}***-pedestal,*** *confirmed by RMP on DIII-D in 2006.*
6. ***2005, August:***

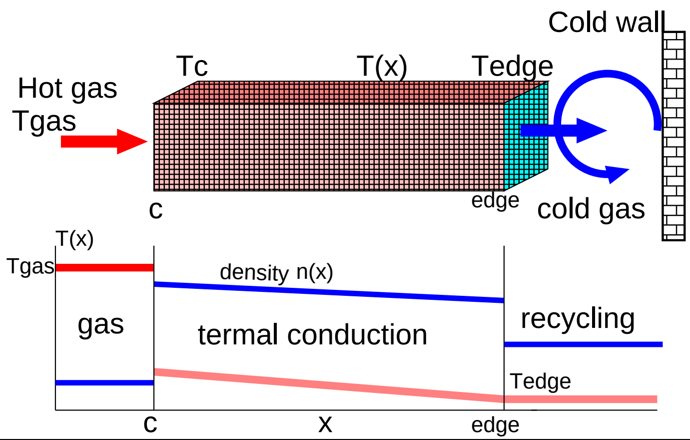
   Stabilization of the plasma edge and ELMs by pumping Li layer was predicted.

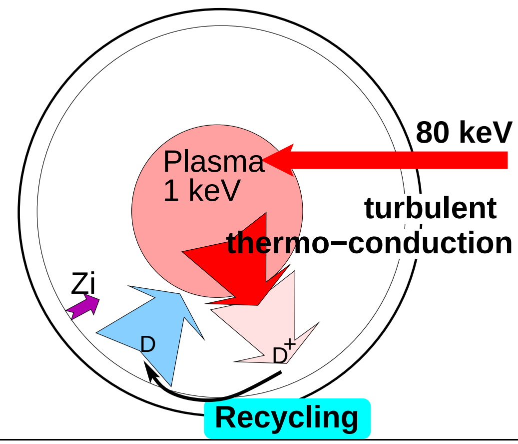
7. ***2006, January:***

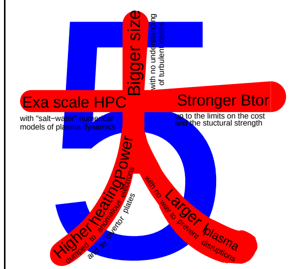
   The properties of new confinement regime was reported to PPPL

2011: The LiWF concept became capable to challenge the entire FES program in all basic plasma physics, technology and reactor related aspects

1.1 Confinement: the mess and the science

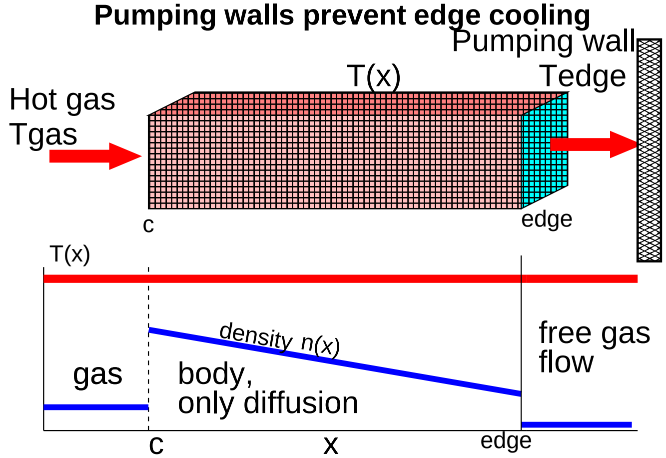


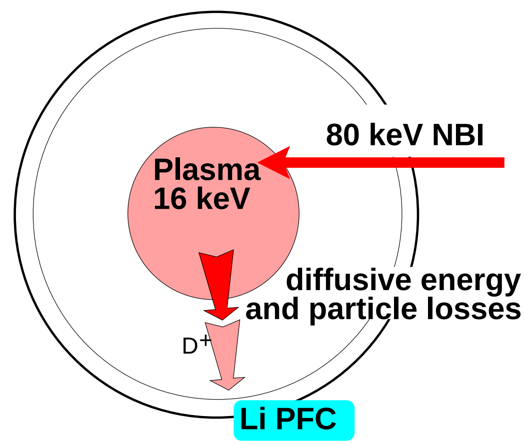




Fusion of 5 "Bigs"

Pumping walls prevent edge cooling





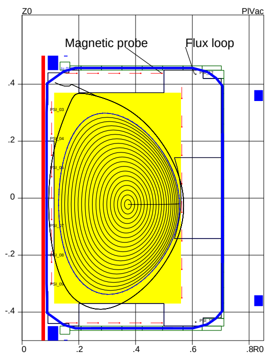
LiWF

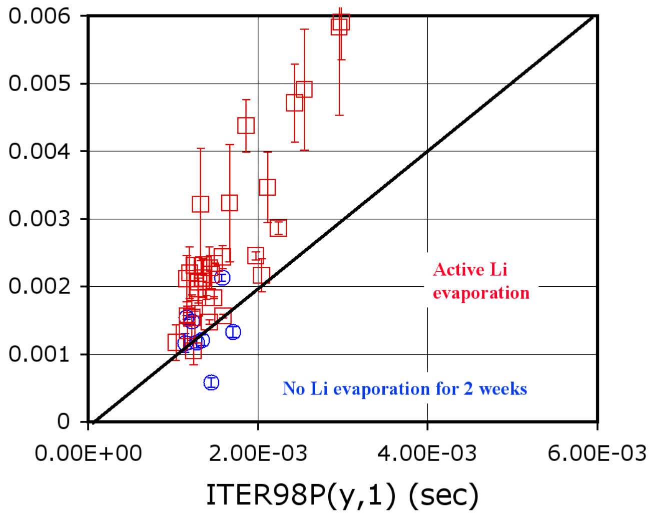
(Lithium Wall Fusion)

high metal χ_e	*high* χ_e *in toroidal plasmas*
modest gas χ_g	*modest ion* χ_i
modest diffusion D_g	*modest plasma diffusion* D_i
"fueling" by gas injection	*NBI* ***fueling*** *of the plasma core*
heating by gas injection	*NBI* ***heating*** *of the plasma core*

Predictions of better confinement confirmed by CDX-U

The pumping Li introduces the best possible confinement regime, which is determined by plasma diffusion





Reference Transport Model

$$\Gamma_{i,e} = \chi_i^{neo-classics} \nabla n \qquad (1.1)$$

easily reproduced global CDX-U parameters (2007)

Parameter	CDX-U	RTM	RTM-0.8	glf23	Comment
$\dot{N}$, 10^{21}part/sec	1-2	.98	0.5	0.8-3	Gas puff rate
β_j	0.160	0.151	0.150	0.145	measured β_j
l_i	0.66	0.769	0.702	0.877	internal induc
V, Volt	0.5-0.6	0.77	0.53	0.85	Loop Voltage
τ_E, msec	3.5-4.5	2.7	3.8	2.3	
$n_e(0)$, $10^{19} part/m^3$		0.9	0.7	0.9	
$T_e(0)$, keV		0.308	0.366	0.329	
$T_i(0)$, keV		0.031	0.029	0.028	

All MHD activity disappeared with Li surface.

Only with after appropriate calibration it was possible to extract the energy confinement time in CDX-U (pulse length 20 msec)

The BEST possible confinement regime was revealed

NBI for core fueling & heating + Pumping LiWall conditions
(Limited plasma edge cooling: $R^{ecycling} < 0.5$, $\Gamma^{gasI} < \Gamma^{NBI}$)

Unique properties:

1. *Energy losses are determined only by particle diffusion*
2. *Anomalous electron thermal conduction plays no role*
3. *Entire plasma volume produces fusion*
4. *The plasma physics is much simpler, no T'-instabilities*
5. *The temperature profile is insensitive to anything except to recycling and NBI energy*

The confinement determined by plasma diffusion makes NBI capable of plasma fueling !

- ***LiWF:*** *The boundary conditions is the major effect in energy confinement. The BEST, diffusion based confinement regime rules out the importance of the core transport.*
- ***FES:*** *Is obsessed with core transport and propaganda of successes in understanding the turbulence with 5D- 6D-simulations. Is totally blind to the plasma boundary conditions (prescribed T_{edge}).*

The entire magnetic fusion was misled by the wrong FES interpretation of the plasma confinement as a core transport problem, resulted in present reliance on big size and heating power of the machines.

LiWF: For toroidal plasma it is much more efficient to prevent plasma cooling by neutrals from the wall than to rely on overwhelming heating power.

1.2 Stability of the plasma edge

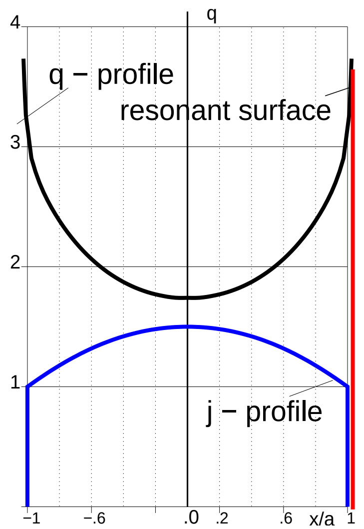
Understood in August 2005, experiment suggested on JET and Tore Supra in Oct. 2005, reported to PPPL on Jan. 11, 2006.

This understanding makes implementation of LiWF regime practical.

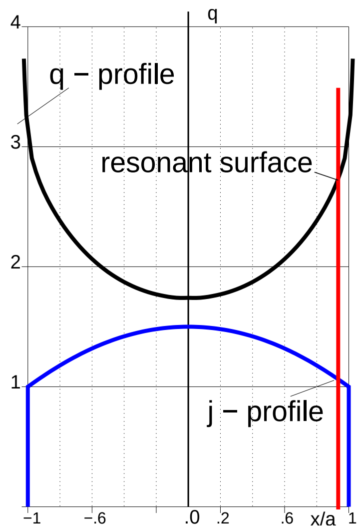
(Lithium covered side walls are not required. Only target plates should be covered by flowing liquid lithium.)

A widespread belief in MHD theory is that the high edge current density is destabilizing

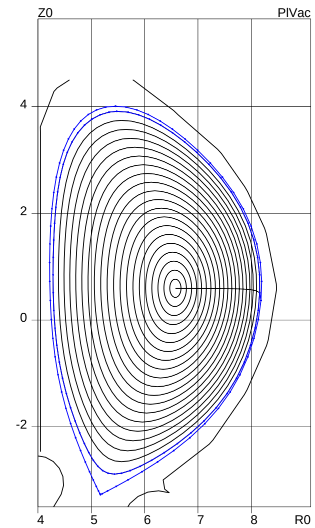
$$W = \frac{\pi}{2\mu_0}\int\left\{r\psi'^2+\frac{m^2}{r}\psi^2+\frac{R\bar{j}'}{B_z(1/q-n/m)}\psi^2\right\}dr \simeq -\frac{\pi}{2\mu_0}\frac{R\bar{j}^{edge}}{B_z(\mu_a-n/m)}\psi^2. \tag{1.2}$$



case 1: $mq_a < n$
Ideally unstable



case 2: $mq_a > n$
Tearing stable



LiWall + Separatrix: $q_a = \infty$
Ideally & tearing stable

In presence of separatrix, the finite edge current density is stabilizing. High T_e^{edge} elevates j^{edge}. Stabilization is confirmed on DIII-D, JET, and on NSTX, EAST, DIII-D with Li.

Stability of LiWF regime vs conventional FES

- ***FES has neither understanding nor ways to prevent plasma disruptions:***

  1. ***The Greenwald limit is involved in operational regimes;***
  2. ***The non-stationary plasma-wall interaction make plasma stability unpredictable;***
  3. ***Tungsten divertor is in conflict with ELMs;***
  4. ***The thermal quench has no precursor;***
  5. ***The high density disruption are prone to runaway electrons, which is impossible to mitigate by MGI;***
  6. ***Big size and large plasma current, necessary to compensate the bad confinement of FES regime, make all engineering problems unsolvable;***

  Regarding macroscopic plasma stability, FES is a collection of mess in theory, ignorance of experimental data and incapable simulation codes.

- ***LiWF suggests the BEST possible stability for a burning plasma***

  1. ***The plasma profiles are predictable and controlled by NBI***
  2. ***Greenwald limit is absent, no ELMs, sawteeth;***
  3. ***The plasma-wall interaction is stationary;***
  4. ***Low-Z Li PFC vs tungsten, with no thermal force driving Li to the plasma;***
  5. ***The global stability is determined by a simple ideal MHD model;***
  6. ***The Runaway electrons are not expected during the current quench;***
  7. ***A practical plasma current of 5 MA is sufficient for DEMO fusion.***

  in 2007 the author of LiWF made a breakthrough in understanding disruptions. With S.Putvinski special experimental measurements: EAST (Hiro currents), Tore Supra, T-10, AUG (repetitive gas injection) were motivated and organized.

1.3 Plasma edge and temperature pedestal

LiWF: from the edge the energy fluxes are transported to the wall by the particle flux:

$$\begin{aligned}
&\frac{5}{2}\Gamma_e^{edge-wall}T_e^{edge} = \underbrace{\int_V P_e dV}_{\substack{heat\ \ source\\ for\ \ electrons}}, \quad T_e^{edge} = \frac{2}{5}\frac{1-R_e^{ecycl}}{\Gamma^{core-edge}+\Gamma^{gasI}}\int_V P_e dV,\\
&\frac{5}{2}\Gamma_i^{edge-wall}T_i^{edge} = \underbrace{\int_V P_i dV}_{\substack{heat\ \ source\\ for\ \ ions}}, \quad T_i^{edge} = \frac{2}{5}\frac{1-R_i^{ecycl}}{\Gamma^{core-edge}+\Gamma^{gasI}}\int_V P_i dV.
\end{aligned} \tag{1.3}$$

These Krasheninnikov's boundary conditions determine $T_{e,i}^{edge}$

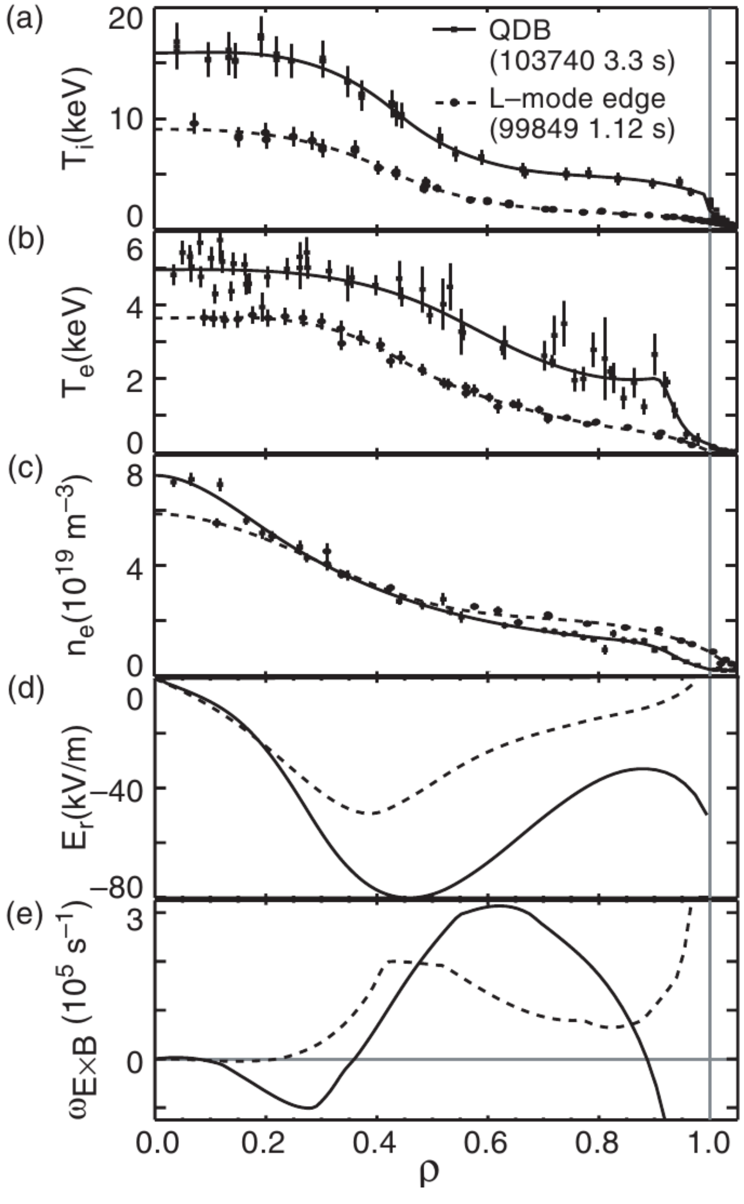
Four effects, determining the edge temperature, are revealed here explicitly:

1. ***Recycling $1 - R^{ecycl}$ (i.e., wall conditions) is the most critical***
2. ***Total heating power $\int PdV$***
3. ***Core confinement $\Gamma^{core-edge}$***
4. ***Outgasing of the walls Γ^{gasI}***

Edge temperature does NOT depend on transport coefficients near the edge.

This property of T^{edge} allows the determination of the position of the plasma edge experimentally

Quiescent H-mode regime on DIII-D



"The quiescent double barrier regime in DIII-D" by C. M. Greenfield, K. H. Burrell, E. J. Doyle et al. Plasma Phys. Control. Fusion 44 (2002) A123-A135.

Figure 4. Kinetic profiles from a QDB (103740) and ITB with an L-mode edge (99849).

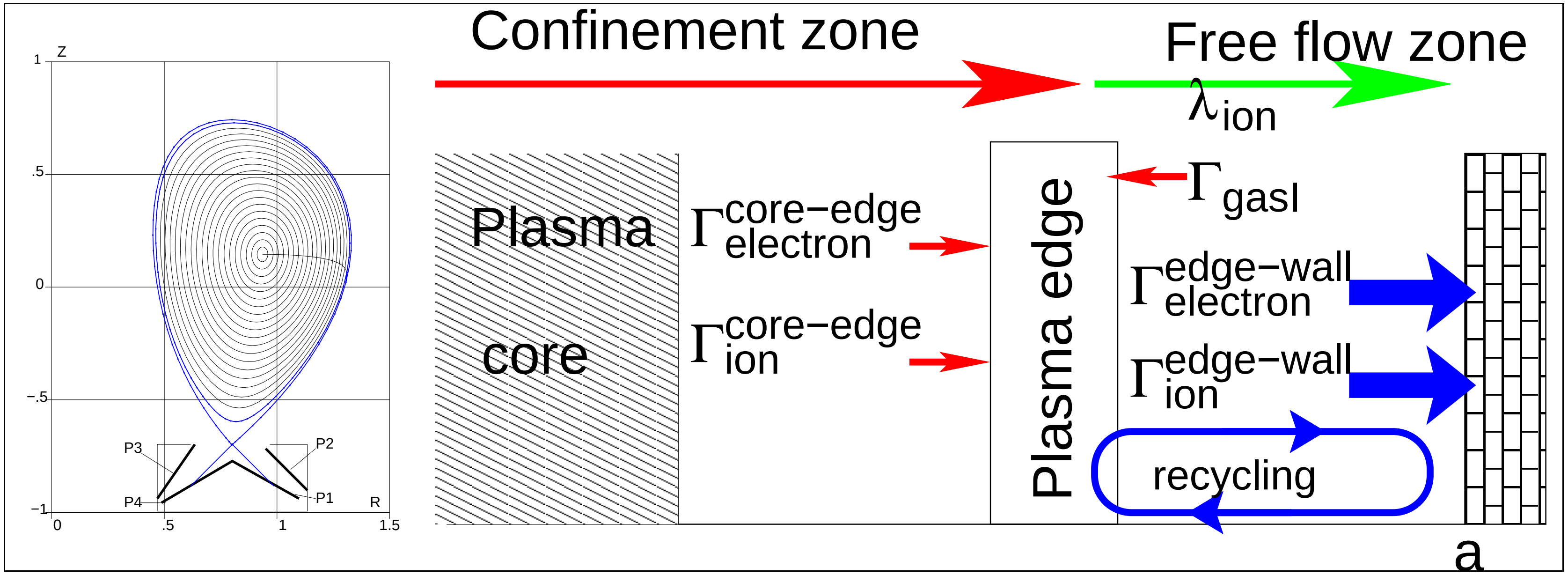
1. ***Ion temperature,***
2. ***Electron temperatures,***
3. ***Electron density,***
4. ***Radial electric field,***
5. ***E x B shearing rate.***

There are many similar pictures from different regimes on DIII-D and from all other machines.

(The record $T_i^{ped} \simeq 6$ kV was achieved in QHM regime.)

Core-transport based interpretation of pedestals

The basic assumption is that the plasma confinement zone (i.e., the core) extends up to the separatrix



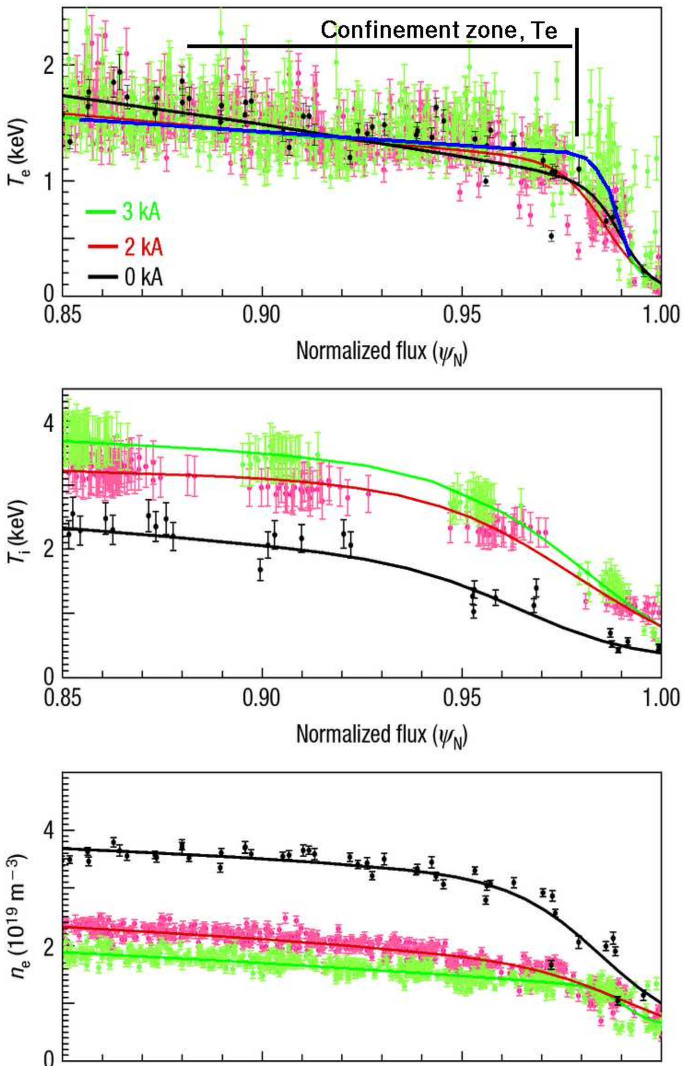
In the core the heat flux

$$q \simeq -n\chi\nabla T. \tag{1.4}$$

Accordingly, the sharp temperature gradient automatically means a reduction of χ down to the neo-classical level and the presence of a transport barrier.

This looks like an unbelievable gift to the plasma physicists from the tokamak edge plasma !

RMP experiments on DIII-D have determined the size of the confinement zone



0 kA, 2 kA, 3 kA $I_{RMP-coil}$

T.Evans at al., Nature physics 2, p.419, (2006)

1. ***The pedestal $T_e^{pedestal}$ is found insensitive to RMP $\rightarrow$ $T_e^{pedestal}$ is the T_e^{edge} $\rightarrow$***
   The tip of the T_e pedestal is the boundary of the confinement zone for electrons.
2. ***RMP do penetrate into the confinement zone:***
   The gradients
   $$n'(x),\ T_e'(x)$$
   in the core are reduced by RMP - indication of "screening".
3. ***Different positions of the "edge" for T_e, T_i, n_e are possible***

Claims about flow shear "stabilization" of turbulence and suppressed transport in the pedestal are baseless.

It is just opposite: there is no electron confinement in the pedestal region.

The pedestal is situated outside the confinement zone

Scrape Off Layer Currents

SOLCs exist even in the most quiet plasma. They are the key to the understanding of the plasma edge.

SOL current and MHD activity in DIII-D

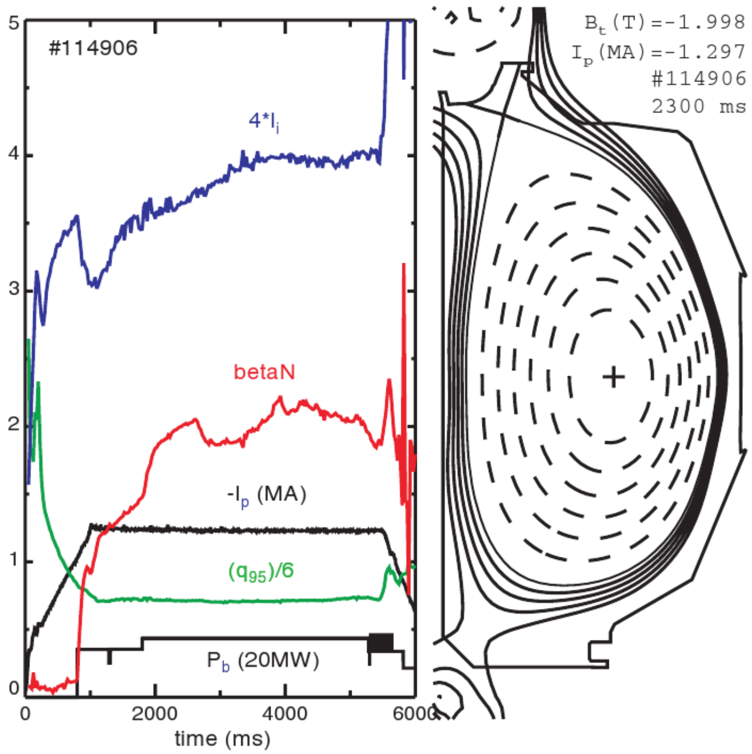


Figure 3. Pictorial discharge summary; the left-hand panel shows I_p in units of megaamperes, P_b in units of 20 MW, q_{95} divided by 6, β_N, and the nominal no-wall limit (here, 4 li). The right-hand panel shows the plasma boundary and four exterior flux surfaces in the

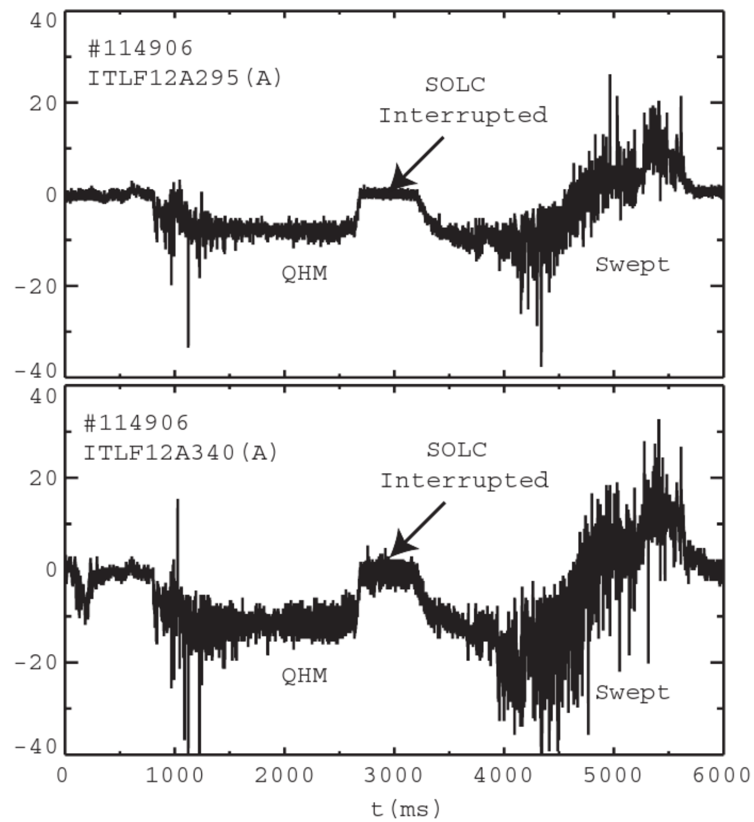
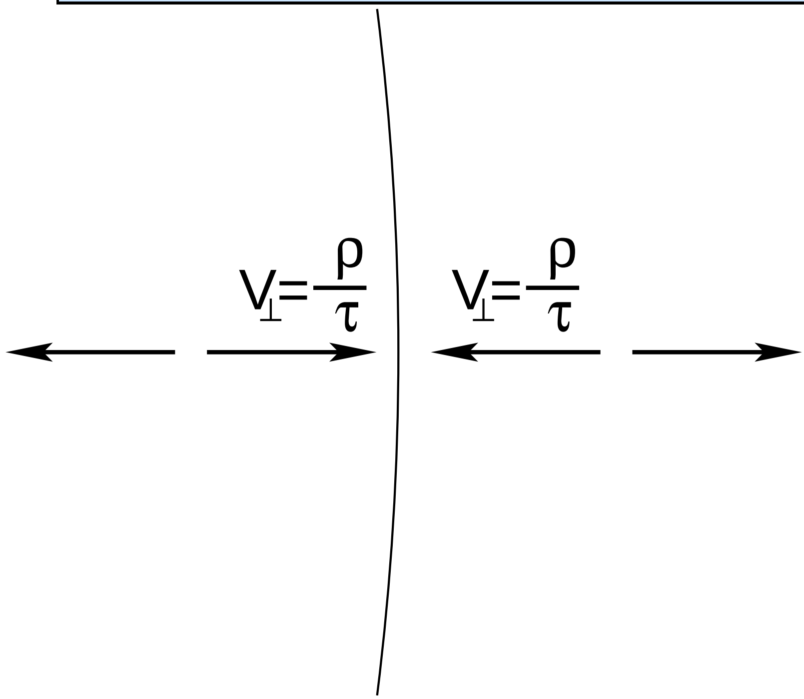


Figure 4. Signals from tile current sensors in tile ring #12 A in the discharge shown in the previous figure. It has a period of QHM over

Todd Evans, Hiro Takahashi and Eric Fredrickson (NF,2004) have found a link between SOLCs and MHD activity on DIII-D. SOLCs are the first candidate for intrinsic perturbations, which determine the width of the temperature pedestal.

Basic mistake in the core-transport based logic

Diffusive transport $\mathrm{q} = n\chi\nabla T$ ***is a result of cancellation of opposit unidirectional fluxes***



$$
\begin{aligned}
\rho^e_{L,cm} &= 0.75 \cdot 10^{-2} \frac{\sqrt{T^e_{keV}}}{B}, \\
\rho^i_{L,cm} &= 0.32 \frac{\sqrt{\mu T^i_{keV}}}{B_T}, \\
\nu_e &= 1.38 \cdot 10^5 \cdot \frac{\Lambda_e}{15} \cdot \frac{n_{20}}{T^{3/2}_{keV}}, \\
\nu_i &= 2580 \cdot \frac{\Lambda_i \sqrt{2}}{17\sqrt{\mu}} \cdot \frac{n_{20}}{T^{3/2}_{keV}}, \\
V^e_{\perp,m/s} &= 10.3 \cdot \frac{\Lambda_e}{15} \cdot \frac{n_{20}}{B_T T_{keV}} \cdot \frac{\rho^e_{orb}}{\rho^e_L}, \\
V^i_{\perp,m/s} &= 8.31 \cdot \frac{\Lambda_i}{17} \cdot \frac{n_{20}}{B_T T_{keV}} \cdot \frac{\rho^e_{orb}}{\rho^e_L}
\end{aligned} \qquad (1.5)
$$

$$
V^e_{\perp,m/s} \simeq V^i_{\perp,m/s} \simeq 10 \frac{n_{20}}{B_T T_{keV}} \cdot \frac{\nu_{eff}}{\nu} \cdot \frac{\rho_{orb}}{\rho_L} \qquad (1.6)
$$

Here, $\rho^{e,i}_{orb}$ ***are the characteristic widths of the electron and ion trajectories, and*** ν_{eff} ***is the effective collision frequency.***

The same Δ determines the thickness of the magnetically perturbed layer which would be sufficient for intercepting the unidirectional fluxes by parallel losses.

Free flow, $L < \lambda_e$:

$$\Delta_{cm} \geq \frac{1}{2} \cdot \frac{L_m}{12.5 \cdot 10^3} \cdot \frac{n_{20}}{B_{p,T} T_{keV}^{3/2}} \cdot \begin{cases} \frac{a}{R} & - \ Phirsh - Schluter \\ \sqrt{2} & - \ Banana \end{cases}, \quad L < \lambda_e. \qquad (1.7)$$

The connection lengths of the order of 12 km (!) are able to destroy the diffusive transport.

Parallel thermal conduction, $L > \lambda_e$: $\chi_{\|} = 3.2 V_{\|} \lambda_e$

$$\Delta_{cm} \geq \frac{1}{2} \cdot \left(\frac{L_m}{3.7 \cdot 10^3}\right)^2 \cdot \frac{n_{20}^2}{B_{p,T} T_{keV}^{7/2}} \cdot \begin{cases} \frac{a}{R} & - \ Phirsh - Schluter \\ \sqrt{2} & - \ Banana \end{cases}, \quad L > \lambda_e. \qquad (1.8)$$

Still the connection lengths of the order of 3.7 km (!) within a region of 1 cm are able to destroy the diffusive transport.

LiWF vs FES regarding plasma edge and pedestal

- ***FES ignores the major effect of perturbed magnetic structure at the edge. Trapped into a baseless assumption of good magnetic surfaces, FES created an ETB-based industry of cooking explanations of plasma phenomena:***

  1. *Shear flow stabilization of turbulence and Suppressed transport to the level below neoclassical in the pedestal region;*
  2. *Screening the external magnetic field perturbations (RMP) by plasma sheared flow;*
  3. *Huge edge localized bootstrap current, "confirmed" by GIGO 5-D kinetic simulations;*
  4. *"Peeling-ballooning" model of ELM stability;*
  5. *EPED model of the width/height of the pedestal.*

  The presence of SoL currents in tokamaks multiplies by zero the value of edge codes like (XGC0,1) with their turbulence and full-f kinetics.

  With absence of a basic understanding, the 5- 6-D codes (e.g., XGC) are used to justify the wrong model and even predict a low plasma edge temperature in the absence of plasma cooling (!).

- ***LiWF provided a science based, predictive understanding of the plasma edge, consistent with experiments:***

  1. *The properties of T_e^{edge} are formulated;*
  2. *The key role of the edge in confinement is clarified;*
  3. *The basic mistake in interpretation of the gradient related transport is revealed;*

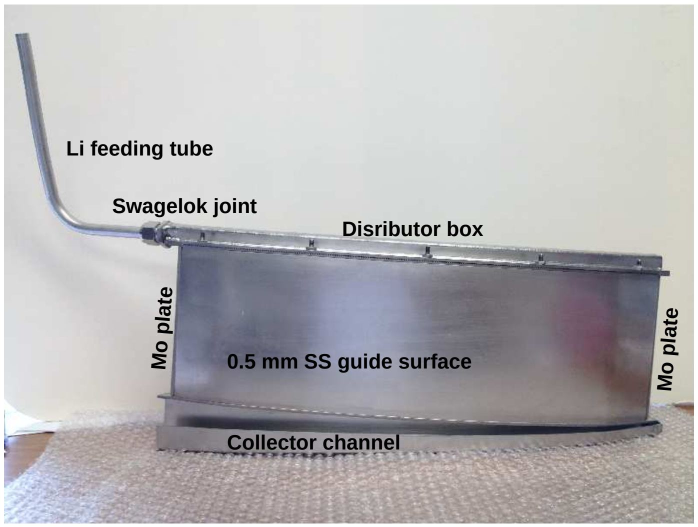
  With LiWF approach to the plasma edge the hopes are created for understanding the L-H transition, the key phenomenon not yet understood.

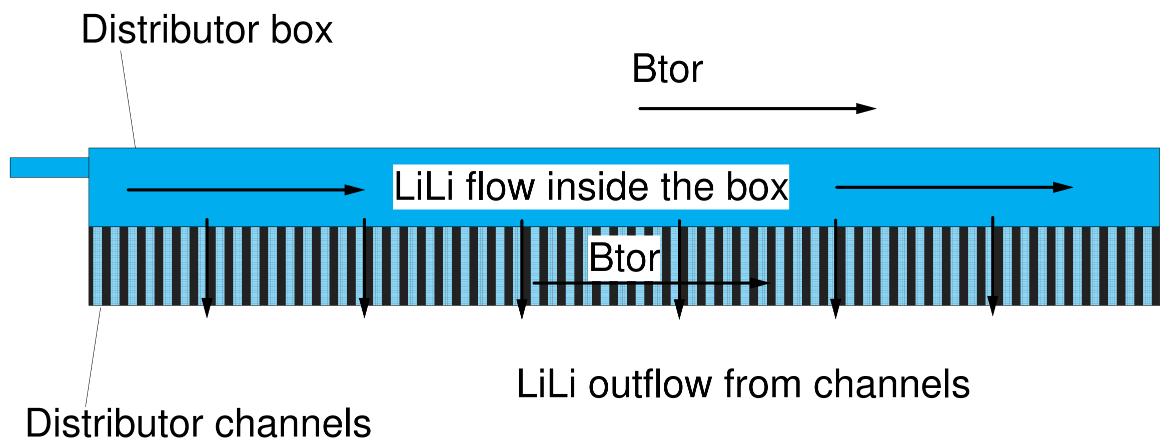
1.4 ${}^{7}_{24}$FLiLi - technology breakthrough (July 2011)

In LiWF during one decade, we understood the tokamak physics much deeper then the entire “Fusion Energy Science” (FES) in the post TFTR era.

ASIPP became a host of LiWF.

Distributor for gravity driven thin viscous flow of LiLi was invented.



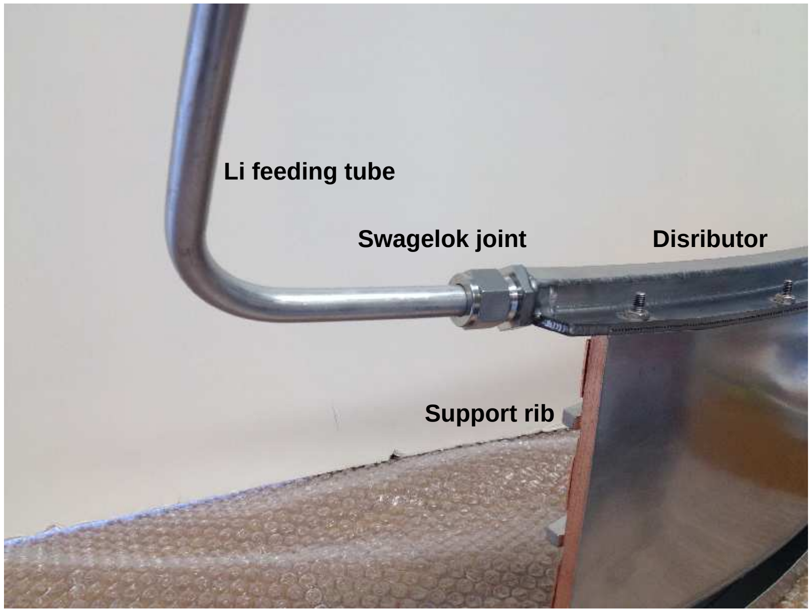


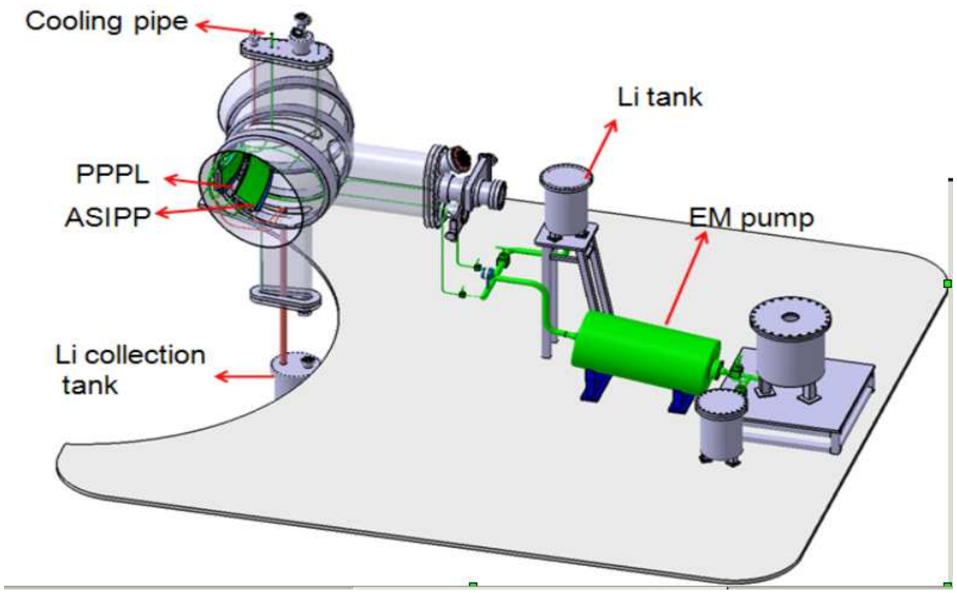
The key design idea, Chang-Lundgren formula

$$\Delta p_{\parallel}^{box} < p^{box} = \Delta p_{\perp}^{channels}, \quad \Delta p \simeq \sigma_E L \bar{V} B^2 \frac{\frac{\sigma_E^{SS}}{\sigma_E} d + 2\delta}{h + \frac{\sigma_E^{SS}}{\sigma_E} d}, \quad \delta \simeq \frac{1.3 \cdot 10^{-2}}{B_T} \textit{[mm]}, \quad (1.9)$$

where L, h *are the length and the size of the flow,* d *is th e thickness of SS walls,* δ *is the Hartmann layer thickness.*

Li feeding pipe and collector





Pressure drop $\Delta p^{FeedPipe}$ along the feeding pipe with inner radius a

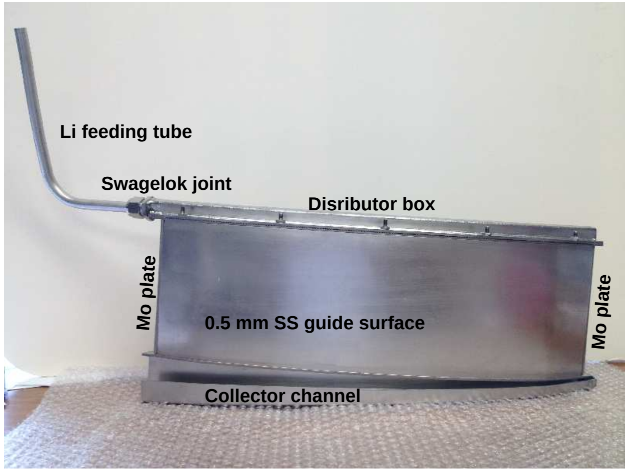
$$\Delta p^{FeedPipe}_{Pa} \simeq \sigma_E \bar{V}^{feed} \frac{\sigma_E^{SS}}{\sigma_E} \frac{d^w}{a} \int_{L^{feed}} (\mathrm{B} \cdot \mathrm{e}_l)^2 dl \simeq 16000 \cdot \bar{V}^{feed} \frac{d^w}{a} \left\langle L_m^{feed} B_\phi^2 \right\rangle . \quad (1.10)$$

Because of MHD effect, the small flow rate of FLiLi can be controlled by the pressure in the Li tank at the level of a fraction of atmosphere

Drain velocity

$$\rho g L^{drain} \simeq 16000 \cdot \bar{V}^{drain}_{cm/s} \frac{d^w}{a} \left\langle L^{drain} B^2_{\phi,T} \right\rangle , \quad \bar{V}^{drain}_{sm/s} = \frac{5}{16} \cdot \frac{a}{d^w} \frac{L^{drain}}{\left\langle L^{drain} B^2_{\phi,T} \right\rangle} . \quad (1.11)$$

Operational FLiLi limiter on HT-7: no separate streams

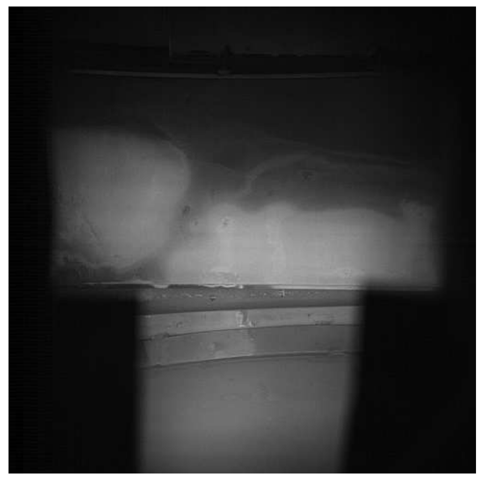


PPPL July 29, 2012

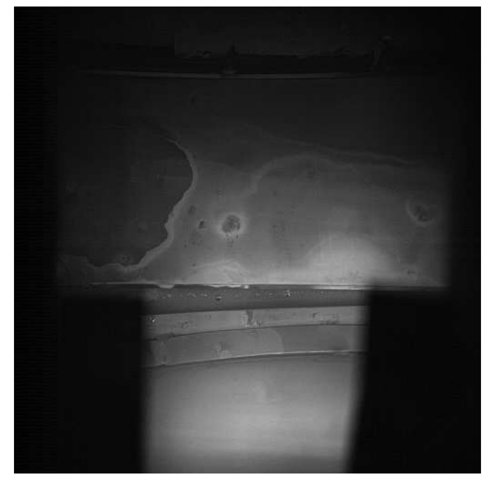
HT-7 ASIPP, August 19, 2012

$P_{Ar} = 15$ **kPa**

$P_{Ar} = 15$ **kPa**



$P_{Ar} = 25$ **kPa**



$P_{Ar} = 40$ **kPa**

Full success with two (PPPL, ASIPP) FLiLi limiters on HT-7, October 4, 2012 (exactly 55 years after the Sputnik launch on Oct. 4, 1957)

Continuous ${}^{7}_{24}$FLiLi systems

${}^{7}_{24}$FLiLi addresses the most fundamental tecnology problem in utilization of LiLi in tokamaks, i.e., the high chemical activity of liquid lithium

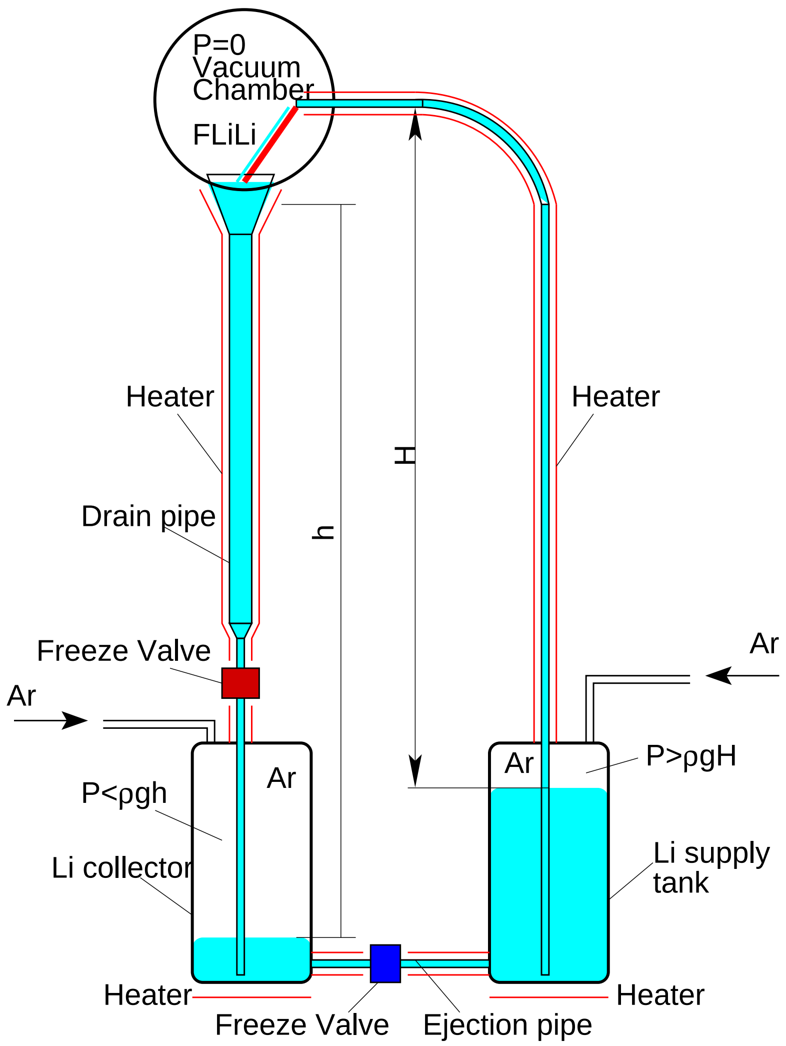
In fact, ${}^{7}_{24}$FLiLi utilizes the chemical activity of LiLi for improvement of in-vessel conditions.

FLiLi should work continuously:

24 hours per day, 7 days per week

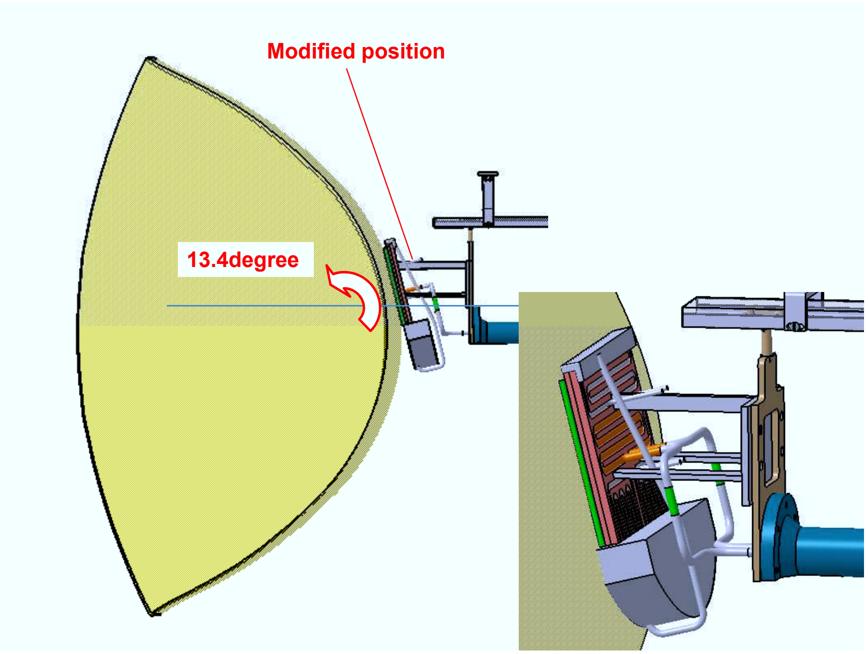
$^{7}_{24}$FLiLi controlled by Ar pressure

No mechanical moving parts in contact with LiLi. Ar pressure $\simeq$ 10s kPa.

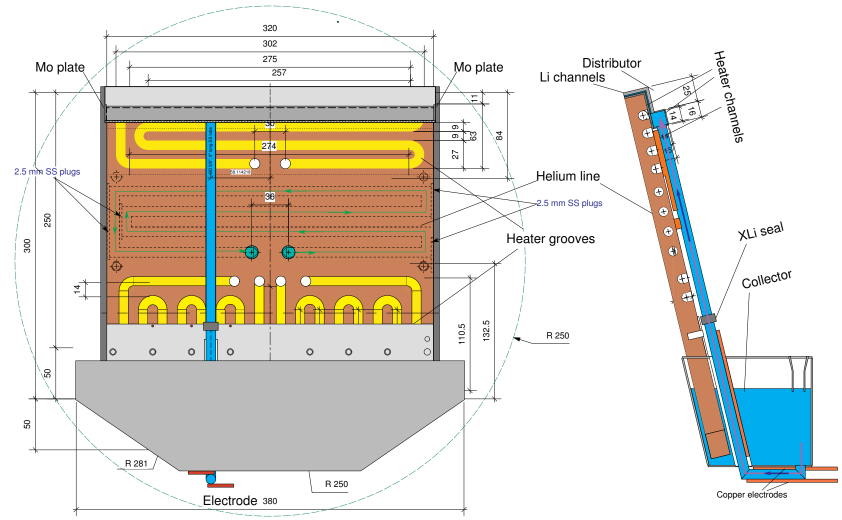


The design concept and a partial implementation

${}^{7}_{24}$FLiLi driven by JxB for EAST (H-port)



FLiLi Limiter inside EAST



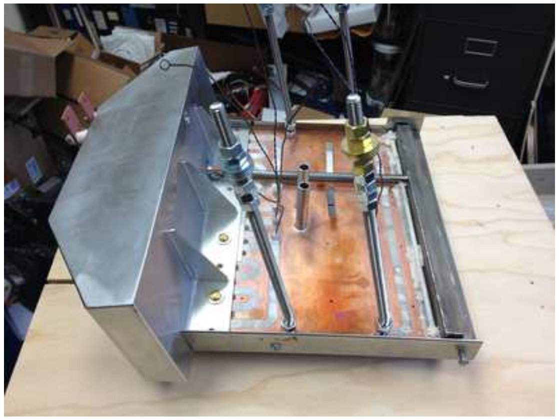
Copper coupon and collector

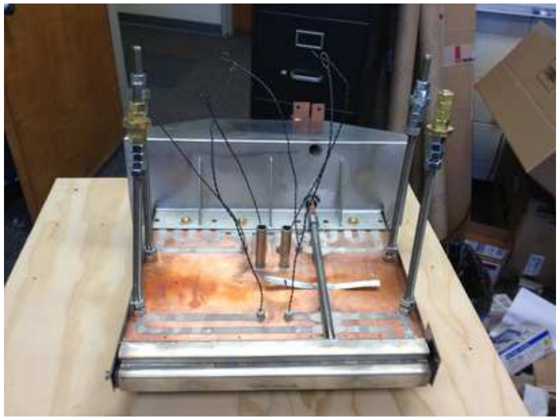
Manufacturing the limiter

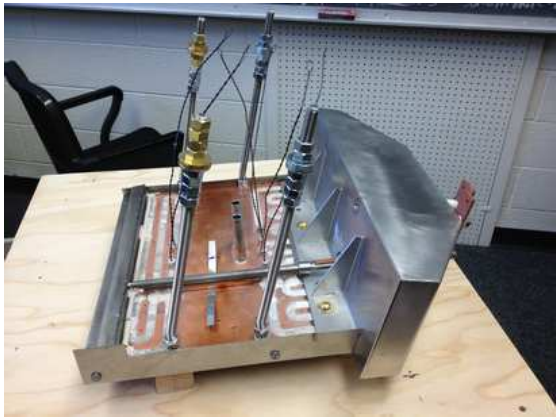
The 0.004" thick SS foil at the front surface of the coupon is an interface layer between LiLi and the copper body.

Welding 0.004" SS foil to the side 0.015" thick SS shim for protection of copper from contact with LiLi is one of manufacturing challenges

Another challenge is to braze the 0.004" SS foil to the copper body







Properties of FLiLi system are simply amazing

In terms of physics:

- ***Scalable in both poloidal and toroidal directions;***
- ***No flow interaction with the tokamak magnetic field (all Hartmann numbers < 1);***
- ***Scalable from a workbench to tokamaks;***
- ***Reliable control of flow rate by the external pressure at the level of fraction of atm;***
- ***Insensitive to the SoL currents at the level of the ion saturation current;***

The physics of FLiLi is clean, simple, easy to analyze and works as theory predicts.

In terms of technology:

- ***The smallest necessary flow rate;***
- ***No high pressure, no mechanical or EM Li pumps;***
- ***Separation of the plasma particle/impurities pumping from power extraction;***

FLiLi resolves the long standing problem of contamination of the Li surface by outgasing from the walls.

In terms of safety:

The inventory of Li with an exposed free surface is minimal (< 0.5 L). It has no ability to create a detonation or ignition level of hydrogen in the vacuum vessel.

1.5 Objectives of the ${}^{7}_{24}$FLiLi project

The near term objectives of this ${}^{7}_{24}$FLiLi proposal is to make a critical contribution to: demonstration of

1. *1000 s long (or stationary) high performance plasma regime on EAST;*
2. *the best possible confinement regimes, determined by the particle diffusion, insensitive to the plasma turbulence, and significantly exceeding the H-mode performance;*
3. *the best stability regimes with elimination of ELMs, density limit, and sawtooth oscillations;*
4. *stationary plasma-wall interactions and the best conditions for the current drive and NBI heating/fueling.*

${}^{7}_{24}$FLiLi limiter installed on EAST.

The next step will be an ${}^{7}_{24}$FLiLi sector of EAST divertor.

The real near term goal is to demonstrate on EAST the feasibility of fusion relevant regimes
and motivate JET management to perform the forthcoming DT experiments in the LiWF regime
with objective of achieving $Q_{DT} > 5$.

1.6 What is the fusion DEMO ?

In conventional fusion there is no valuable DEMO concept.

The 100-200 MW FFRF of the LiWF with its innovative burning plasma regime is the first realistic model of DEMO. It has both fusion and fusion-fission missions

2011, вып. 3 Fusion-fission research facility (FFRF) as a practical step toward hybrids

УДК 621.039.6

FUSION-FISSION RESEARCH FACILITY (FFRF) AS A PRACTICAL STEP TOWARD HYBRIDS

L.E. Zakharov (Princeton University, Princeton Plasma Physics Laboratory, MS-27, P.O. Box 451, Princeton, New Jersey 08543), J. Li, Y. Wu (Institute of Plasma Physics, Chinese Academy of Sciences P.O.Box 1126, Hefei, Anhui 230031, China)

This is your DEMO
LZ

The paper outlines the design of the Fusion-Fission Research Facility (FFRF) for study of hybrid reactors with a unique fusion mission and a pioneering mission of merging fusion and fission physics and technology for accumulation of engineering, physical, and operational data for future hybrid applications. The design was developed in the Institute of Plasma Physics of the Chinese Academy of Sciences (ASIPP) with Princeton Plasma Physics Laboratory (USA) participation. Parameters of the facility are R/a = 4/1m/m, I_{pl} = 5 MA, B_{tor} = 4—6 T, P^{DT} = 50—100 MW, $P^{fission}$ = 80—4000 MW, 1 m thick blanket. The design of FFRF will use as much as possible the EAST and ITER design experience. On the other hand, FFRF strongly relies on new, Lithium Wall Fusion plasma regimes, the development of which has already started in the US and China. The calculations presented in the paper show the large potential of FFRF as a neutron source for driving the fission blanket and developing the fusion-fission technology and applications. The design of the tokamak core does not represent significant challenges and can already proceed to the conceptual design phase.

Keywords: fusion power, tokamak, fusion-fission hybrids, magnetic confinement, burning plasma.

ИССЛЕДОВАТЕЛЬСКАЯ УСТАНОВКА СИНТЕЗА-ДЕЛЕНИЯ FFRF — ПРАКТИЧЕСКИЙ ШАГ К ГИБРИДНЫМ РЕАКТОРАМ. Л.Е. ЗАХАРОВ, J. LI, Y. WU. Статья описывает проект установки FFRF (Fusion-Fission Research Facility) для исследований гибридных реакторов с уникальной миссией в термоядерном синтезе и пионерской миссией соединения термоядерной и ядерной физики и технологии для получения инженерных, физических и эксплуатационных данных для будущих применений гибридных технологий. Проект установки разработан в Институте физики плазмы Китайской академии наук (при участии Принстонской лаборатории физики плазмы, США) с параметрами: R/a = 4/1, м/м, I_{pl} = 5 MA, B_{tor} = 4—6 Тл, P^{DT} = 50—100 МВт, $P^{fission}$ = 80—4000 МВт, ширина бланкета 1 м. Проектирование FFRF будет в полной мере использовать опыт установок EAST и ИТЭР. В то же время FFRF существенно опирается на новые плазменные режимы с литиевой стенкой, разработка которых уже началась в США и Китае. Представленные в статье расчёты показывают большие возможности FFRF как нейтронного источника для «питания» нейтронами бланкета деления и разработки гибридной (синтеза-деления) технологии и её применений. Проектирование основных компонентов токамака не требует решения очень сложных проблем. Поэтому концептуальное проектирование установки может быть начато в ближайшее время.

Ключевые слова: термоядерная энергия, токамаки, ядерно-термоядерные гибриды, магнитное удержание плазмы, термоядерная реакция в плазме.

MISSION AND SCIENTIFIC STRATEGY

FFRF is a project of ASIPP, which is an institution working on the development of fusion applications to nuclear energy.

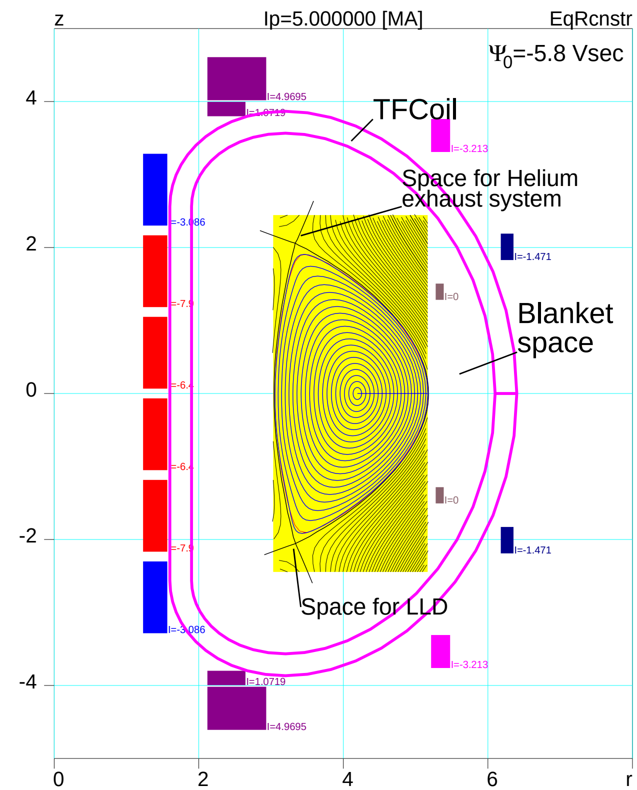
The mission of FFRF is to advance fusion to the level of a stationary neutron source and to create a

On the left is my recommendation to Jiangang Li on the concept for the next-step (two) DEMO devices in China

Two similar devices, DEMO-D (no tritium) and DEMO-T (with DT power) are necessary, in order to assure the success and resolution of potential operational problems in activated DEMO-T.

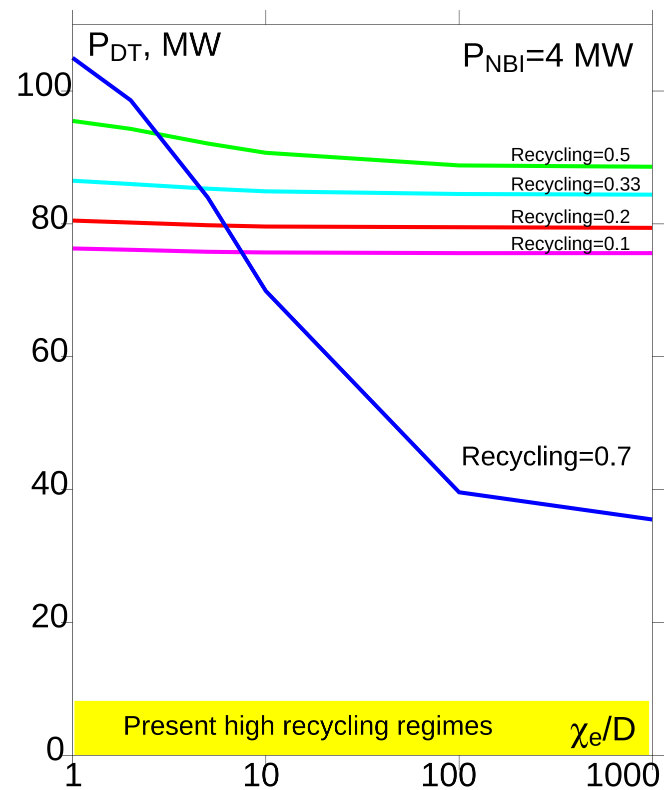
DEMO: mission, parameters and burning plasma

LiWF suggests a realistic, science based DEMO for burning plasma



Parameter	FFRF
$d_{blanket,m}$	1
a_m, R_m	1.0, 4.0
$V^{pl}_{m^3}, S^{pl}_{m^2}$	130, 230
n_{20}	0.4
E^{NBI}_{keV}	120
$\frac{T_i+T_e}{2}\|_{keV}$	24-27
$B_{t,T}$	4-6
$I_{pl,MA}$	5
$\Delta\Psi_{f-top,Vsec}$	40
$\Delta t^{inductive}_{f-top,s}$	$>$4000
$W_{th,MJ}$	42
$\tau^{ind}_{E,sec}$	20-7
P^{NBI}_{MW}	2-5
P^{DT}_{MW}	70-100

Active fission core power 80-4000 MW.



P^{DT} vs electron anomaly

At the practical level of $R^{ecycl} < 0.5$, the burning plasma regime with $P^{DT} = 50 - 100$ MW is possible in FFRF

Remarkably, a robust "hot-ion" regime was found (thanks to G.Hammett) where the cyclotron radiation keeps $T_e < T_i$ even with the α-particle heating of electrons.

LiWF DEMO vs FES

- ***FES is incapable to create a practical concept for the next step burning plasma device:***

  1. *It has no even definition of DEMO;*
  2. *What FES suggests as a DEMO version is simply ridiculous from the physics point of view;*
  3. *The failure of ITER, the product of FES, with tungsten divertor will end any speculations about DEMO;*

- ***LiWF provided a constructive concept of DEMO and the path to it:***

  1. *Development and demonstration of the LiWF regime on EAST;*
  2. *Motivating JET management to perform the forthcoming DT experiments in the LiWF regime with objective of achieving Q$_{DT}$ $>$5.*
  3. *Move toward the design of 100-200 MW DEMO, based on LiWF regime.*

LiWF is unique in creating the realistic near and middle term vision of magnetic fusion power development.

In a burning plasma, the problems of plasma physics and technology cannot be considered as a set of separated "bricks" of challenges.

All problems are interlinked.

They can be addressed only in a self-consistent manner.

The power extraction is an example

1. ***Protection of PFC from the plasma:***
   - ***Solid PFC can reliably operate only at $\simeq 5$ MW/m^2, what is highly insufficient.***
   - ***Reliance on the plasma physics (radiation, detachment) with the present lack of basic plasma understanding is a worse choice.***
   - ***ELMs erode the solid PFC regardless of design efforts.***
2. ***Protection of the plasma from the PFC is a much more challenging problem:***
   - ***All high-Z materials, solid and liquid, are incompatible with high T$_{edge}$.***
   - ***Li cannot be used for power extraction due to low thermal conduction and easy evaporation.***
   - ***The tokamak plasma is sensitive to inconsistencies. It responds by disruptions.***

New regimes is a practical approach to the problem

FES suggests a reliance on improvement of materials for PFC. No hopes with solid materials.

By material development it is impossible to enhance the power extraction even by a factor 2.

A recent innovation in evaporating 18 (!) g/s of Li to the chamber per each 10 MW heat flux from the plasma can be qualified only as a non-sense.

There is no solution based solely on material development !

The only solution is in development of LiWF fusion relevant plasma regimes with order of magnitude better confinement, the best possible stability (no ELMs), steady plasma-wall interactions.

An order of magnitude, rather than a factor 2, enhancement of the energy confinement is a real potential of the innovative tokamak regimes for a burning plasma.

For more than 4 decades the inertia term in the equation of motion

$$\rho\frac{d\vec{V}}{dt} = -\nabla p + (\vec{j}\times\vec{B}), \tag{2.1}$$

represents an unresolved obstacle for simulations of tokamaks: fast magneto-sonic waves, which play no role in tokamaks, still require a very small time step.

In fact, the macroscopic tokamak plasma dynamics is driven by a small imbalance of big forces, which are much bigger than the plasma inertia

$$\tau_{MHD} \simeq \frac{R}{V_A} = \underbrace{\frac{R}{2.18\cdot 10^6 B}}_{<1\ \mu s} \quad \ll \quad \underbrace{\tau_{TMHD}}_{\simeq 1\ ms} \quad \ll \quad \underbrace{\tau_{transport}}_{\simeq 0.1\ s} \quad \ll \quad \underbrace{\tau_{resistive}}_{\simeq 1\ s} \tag{2.2}$$

Inertial MHD is not the case of tokamaks !

The macroscopic TMHD is, in fact, a fast equilibrium evolution with excitation of sheet currents or islands at the resonant surfaces and surface currents at the plasma boundary due to magnetic flux conservation ($\tau_{TMHD} \ll \tau_{resistive}$)

In TMHD, following Kadomtsev and Pogutse (1973), the plasma inertia is replaced by a displacement term, which is equivalent to a friction force $\propto -\vec{V}$:

$$\lambda\vec{\xi} = -\nabla p + (\vec{j}\times\vec{B}), \quad \lambda\vec{\xi} \equiv \gamma\vec{V}. \tag{2.3}$$

This replacement provides an iteration algorithm for driving the system. By eliminating plasma oscillations it removes the 4-decade old problem with Courant limitations on the time step in MHD simulations.

Plasma anisotropy and adaptive grids

Originally, the term MHD was associated with dynamics of liquid metals in a magnetic field.
Two important properties of the tokamak plasma make TMHD different from MHD of liquid metals:

1. ***very high plasma anisotropy***

$$(\vec{B} \cdot \nabla T_e) \simeq 0 \quad \rightarrow (\vec{B} \cdot \nabla \sigma) = 0 \tag{2.4}$$

($\sigma = \sigma(T_e)$ is the electric conductivity), and

2. ***absence of restrictions on the plasma flow to the wall***

$$V_{normal} \neq 0 \tag{2.5}$$

(In terms of interaction with the wall, the tokamak plasma cannot be represented by fluid model)

Liquid metals in magnetic field CAN be simulated using hydro-dynamic numerical schemes of NIMROD, M3D, JOREK, etc, properly adapted.

The tokamak plasma cannot be simulated by NIMROD, M3D, JOREK, etc, which ignore the basic physics of plasma-wall interaction

In tokamaks

$$V_{plasma\ to\ the\ wall} \neq 0$$

2.1 TMHD equations

The simplest form of macroscopic TMHD is represented by

$$\lambda\vec{\xi} = -\nabla p + (\vec{j}\times\vec{B}), \quad \vec{B} = (\nabla\times\vec{A}), \quad \mu_0\vec{j} = (\nabla\times\vec{B}),$$
$$-\frac{\partial\vec{A}}{\partial t} - \nabla\varphi_E + (\vec{V}\times\vec{B}) = \frac{\vec{j}}{\sigma}, \quad (\vec{B}\cdot\nabla\sigma) = 0 \tag{2.6}$$
$$\vec{V} \equiv \frac{\partial\vec{\xi}}{\partial t}, \quad (\nabla\cdot\vec{V}) = 0 \qquad \textit{(replaces the equation of state)},$$

The blue terms in (2.6) reflect the distinction between TMHD and conventional MHD:

(a) inertia term is replaced by an effective "friction", (b) plasma anisotropy is expressed explicitly, (c) the plasma velocity is the secondary variable.

Also, plasma ions hitting the wall are converted into neutral atoms, not participating in MHD dynamics.

As a result, (d) plasma is allowed to flow into the wall with no special restrictions

The hydrodynamic boundary condition $V_{normal} = 0$ at the wall

adopted so far in all 3-D MHD codes

does not exist in TMHD

New numerical schemes, based on adaptive coordinate systems aligned with magnetic field are required by specifics of TMHD.

The equilibrium equations can be derived from a variational principle

(V.D. Shafranov (Voprosy Teorii Plasmy, GosAtomIsdat, v .2, 1963, (Reviews of Plasma Physics, Vol. 2, 1966, p. 103)))

$$W_{TMHD} = \frac{1}{\mu_0}\int\left\{\lambda\mu_0\frac{\vec{\xi}^2}{2} + \frac{\mathrm{B}^2}{2} - \bar{p}\right\}dv, \quad \bar{p} \equiv \mu_0 p, \tag{2.7}$$

$$\delta\mathrm{B} = (\nabla\times(\vec{\xi}\times\mathrm{B})), \quad \delta\bar{p} = -(\vec{\xi}\cdot\nabla\bar{p})$$

with $\vec{\xi}$ as a variation.

Numerically, approximated by finite elements, the minimization of this functional leads to symmetric matrix equations.

As a result, the TMHD equations can be solved using Cholesky decomposition!!!

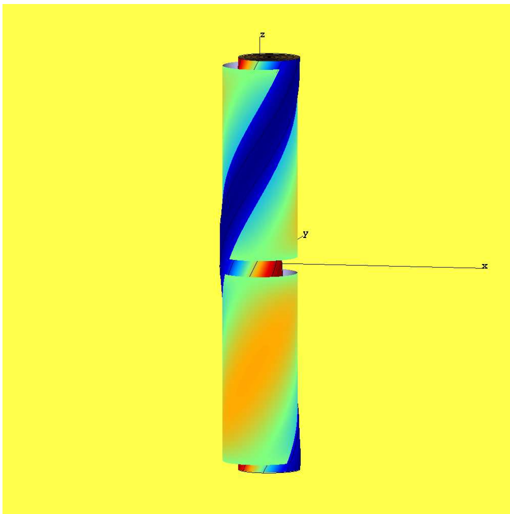
It is well suitable for GPU !

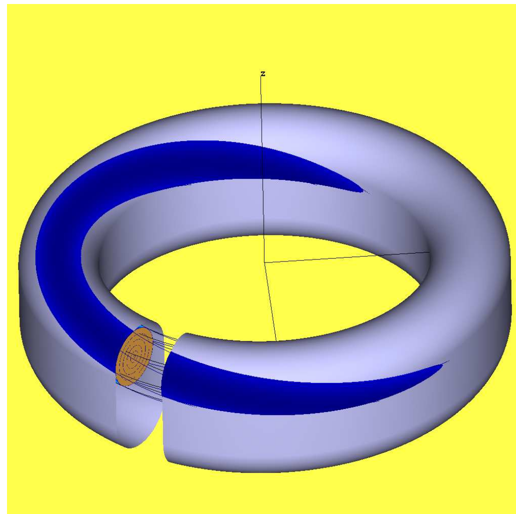
The applications of Cholesky decomposition is given below for 2-D equilibria.

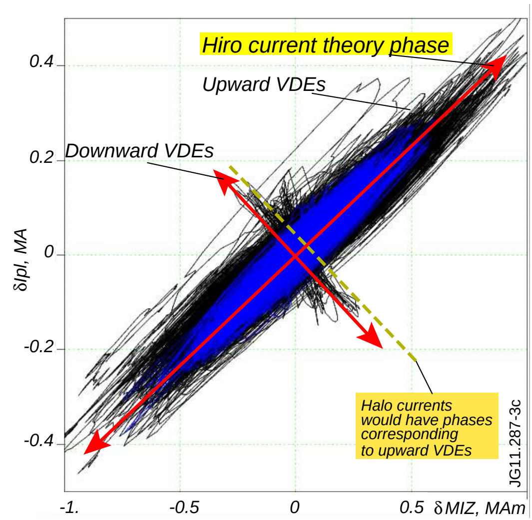
Recently with Xujing Li we understood the use of GPU and created a collisional particle orbit routine which for 80,000 orbits is 40 times faster than the 32 processor CPU.

2.2 Hiro currents and Wall Touching Kink Mode (WTKM)

Toroidal asymmetry in the plasma current measurements during VDE on JET was explained in 2007 by the theory of the Wall Touching Kink Mode. Its Hiro currents are responsible for asymmetry. The halo currents would lead to the opposit sign of the effect.







As a side result, the "salt-water" boundary condition $V_{normal} = 0$, irrelevant to the tokamak plasma, was revealed in all 3-D MHD codes.

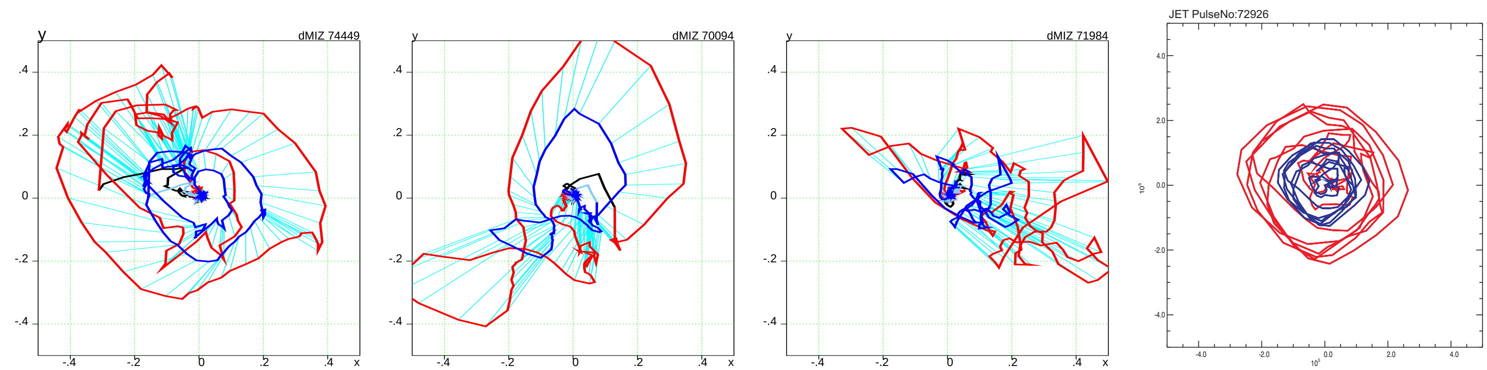
The theory of WTKM multiplies by zero the applicability of the 3-D MHD codes (M3D, NIMROD) for disruption simulations.

Important properties of Hiro currents

- *Absolutely necessary for slowing down plasma dynamics from μs-time scales to observable $1-10ms$ equilibrium evolution;*
- *Plasma act as Hiro current, rather than voltage generator;*
- *Can confuse interpretations of magnetic measurement regarding plasma displacement, values of a, q_a;*
- *Can shorten the gaps between tiles and create large electric circuits along the PFC surface (in contrast to broken by gaps eddy currents);*
- *Can significantly affect plasma azimuthal motion and rotation.*

Theory confirmed the early (2007) assessment of Fx forces in ITER by JET engineers, thus, making the issue addressed.

Understanding of the disruption physics is impossible without understanding effects associated with Hiro currents. Mode rotation is an important challenge.



Red line: $\delta\vec{M}_{iz}(t) = \delta M_{iz,5-1}(t)\vec{e}_x + \delta M_{iz,7-3}(t)\vec{e}_y$

Blue line: $\delta\vec{I}_{pl}(t) = \delta I_{pl,5-1}(t)\vec{e}_x + \delta I_{pl,7-3}(t)\vec{e}_y$

2.3 2-D version of DSC simulates 3 regimes of WTKM

The Wall Touching Kink Mode (WTKM), associated with the Hiro currents, is a new kind of MHD modes. It is well distinguishable from the Free Boundary Kink Modes (FBKM).

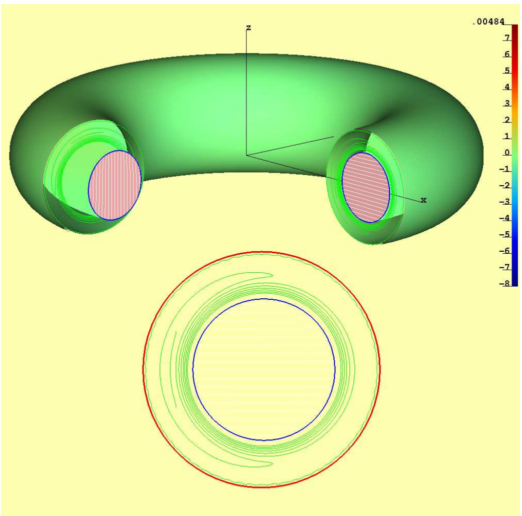
WTKM is a natural candidate for triggering the thermal quench (HL-2A has data.)

New codes, based on adaptive grids are necessary for simulation of WTKM.

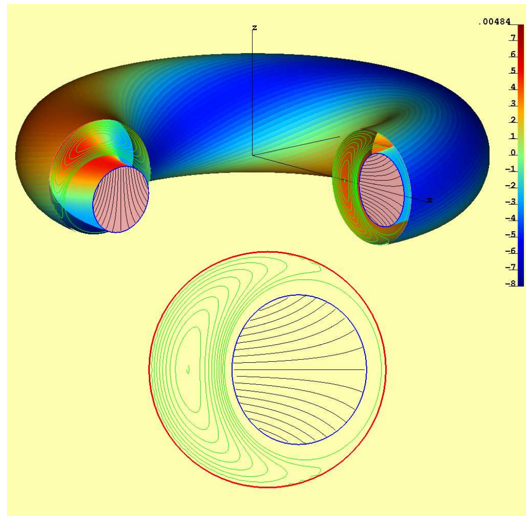
So far, Kadomsev-Pogutse reduced MHD model was implemented.

Movie 1: Free boundary kink mode inside a shell

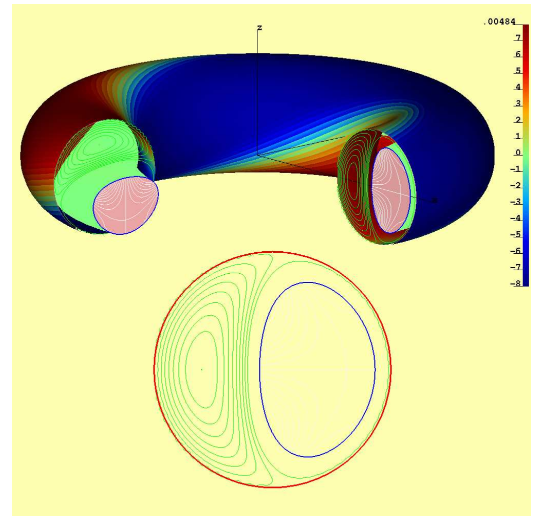
Fast regime of the kink mode inside the ideal wall (idealized theoretical model)



Initial perturbed plasma



Fast phase of instability



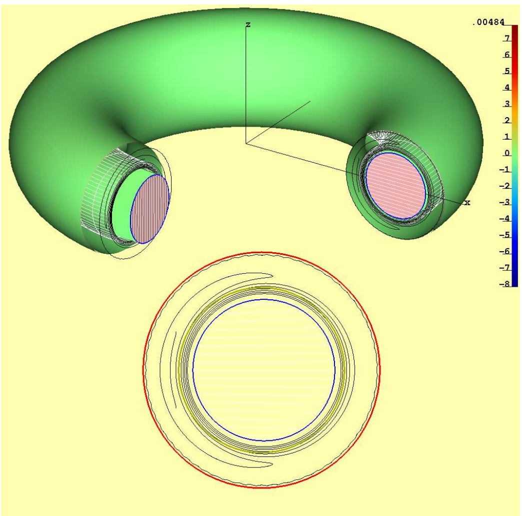
Saturation of the mode

After saturation, plasma is maintained in equilibrium by the eddy currents in the ideal wall.

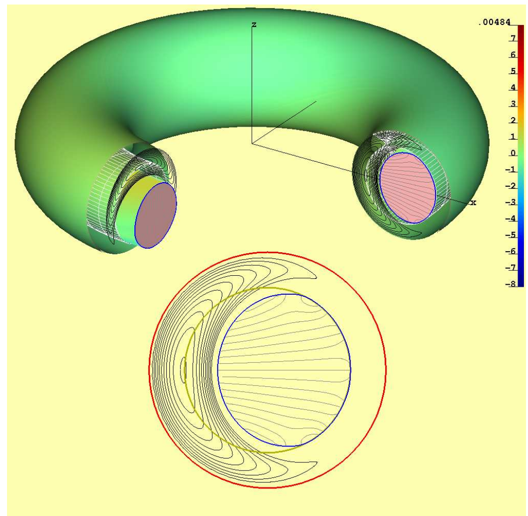
NIMROD can simulate this regime

Movie 2: Wall touching kink mode. Hiro current excitation

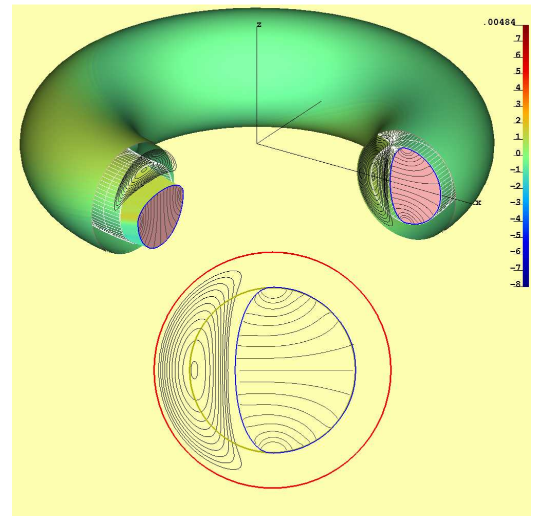
Fast regime of the wall touching kink mode inside the tile surface



Initial perturbed plasma



Fast phase of instability, excitation of Hiro currents



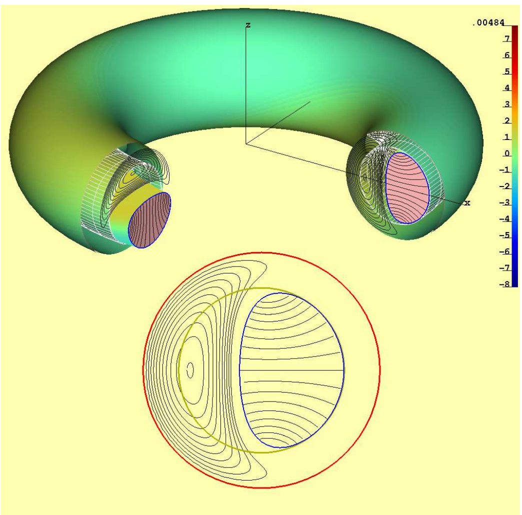
Saturation of the mode due to Hiro currents

Plasma motion slows down due to excitation of the Hiro currents along the tile surface.

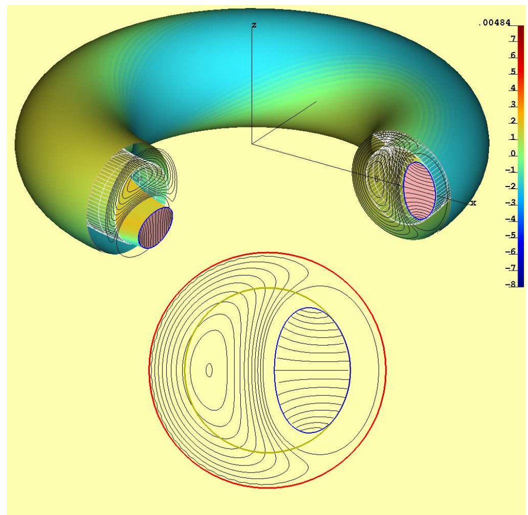
Movie 3: Plasma decay and termination

Self-consistent plasma/(Hiro currents) decay with plasma moving into the wall.

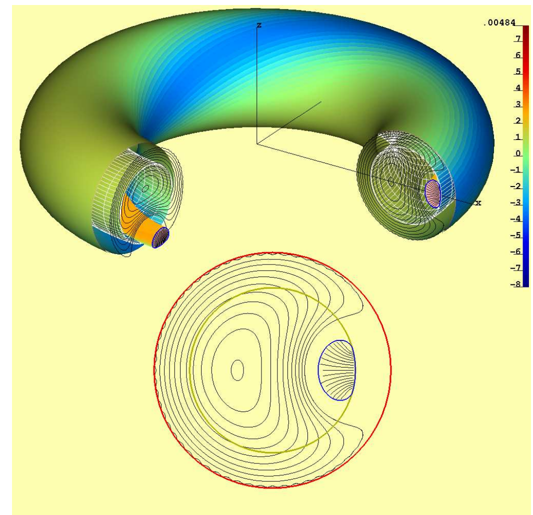
This is the most important new regime for MHD simulations.



Initial phase of decay



Intermediate phase of decay



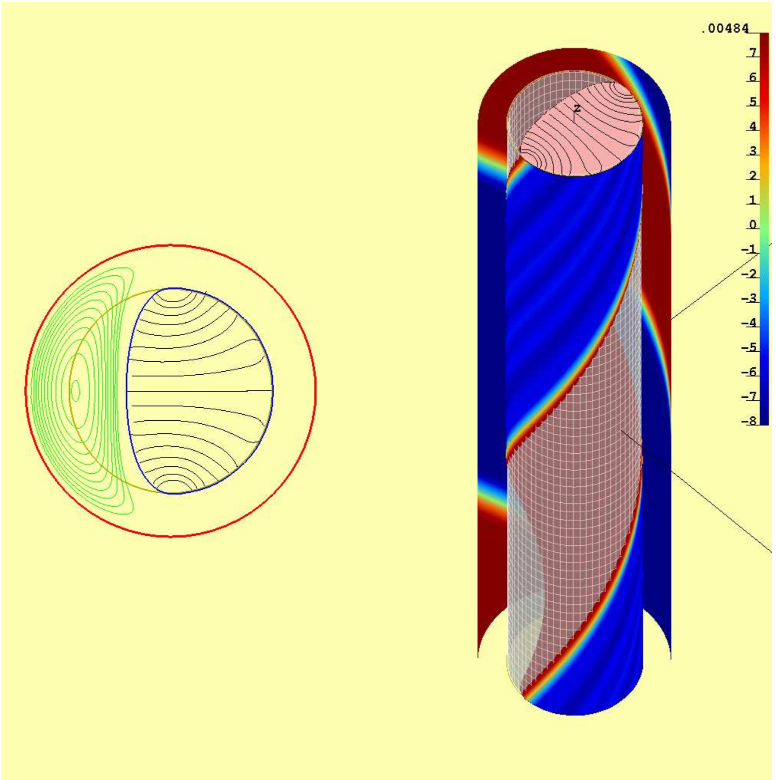
Final phase of plasma termination

Two regimes: (a) generation of the Hiro currents, and (b) plasma decay cannot be reproduced by existing 3-D numerical codes

"Salt water" boundary condition $V_{normal} = 0$, remaining uncorrected for already 6.5 (!) years, makes M3D, NIMROD and JOREK codes irrelevant to disruption simulations

2.4 VDE, Hiro, Evans and “halo” currents

During instability fast plasma motion is stopped by the Hiro currents in tiles



Transient equilibrium maintained by Hiro currents

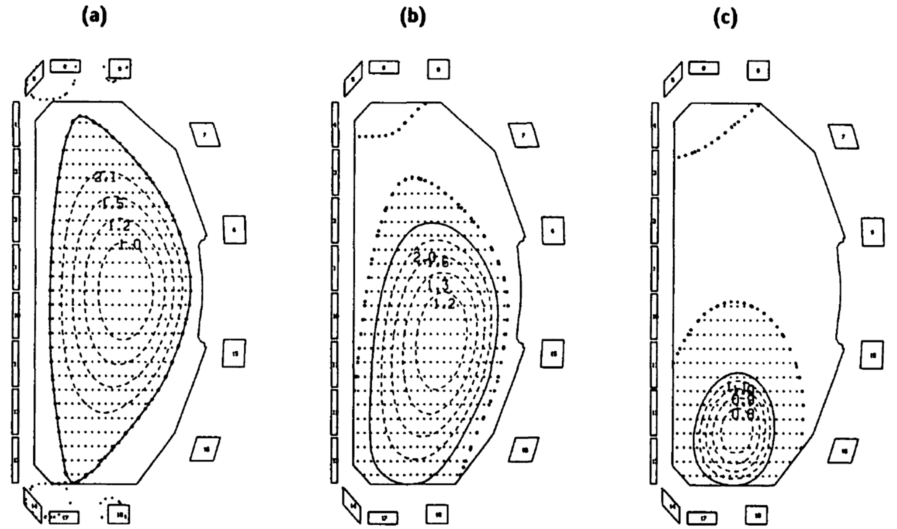
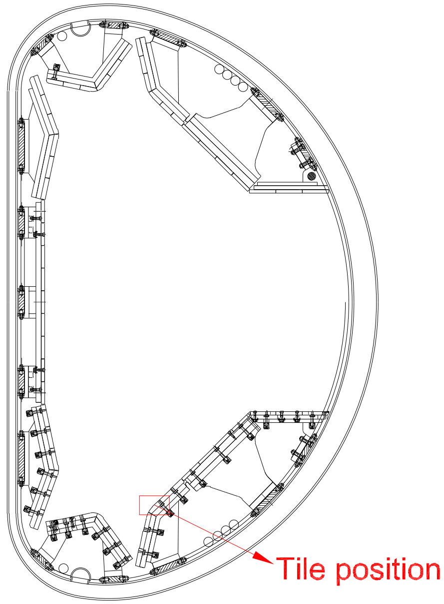


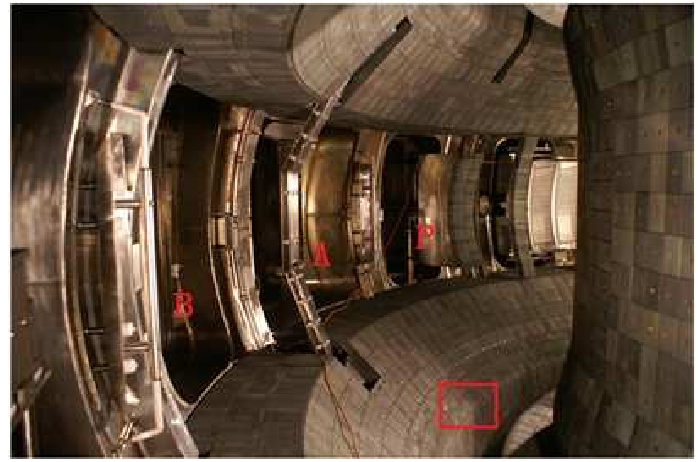
FIG. 3. Equilibrium flux plots from EFIT at three times during the vertical instability: (a) 2660 ms, (b) 2675 ms and (c) 2684 ms. Plasma current was allowed in the hatched region, including part of the SOL.

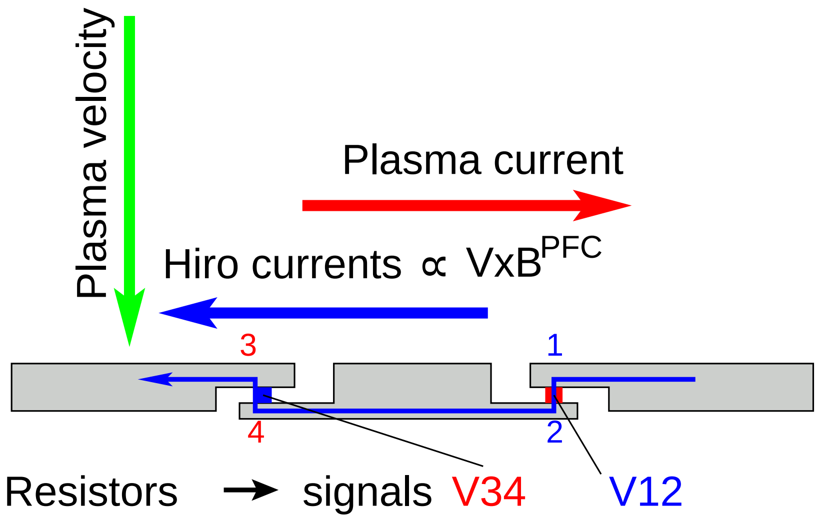
VDE tile currents suggest totally different interpretation.

- ***Negative Hiro currents are flowing along the tile surface***
- ***Positive (force free) surface currents from the plasma edge may go to the tile surface as “Evans” currents. They are measured, but misinterpreted as the halo currents.***

The certified MHD experts of FES make all efforts to devaluate the Hiro currents and, at least, to blend them with “halo” currents.



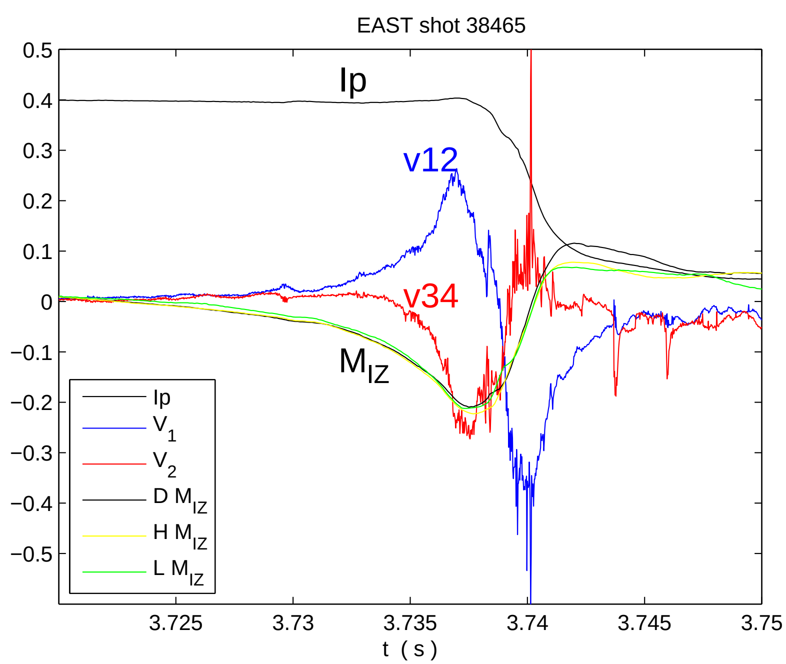




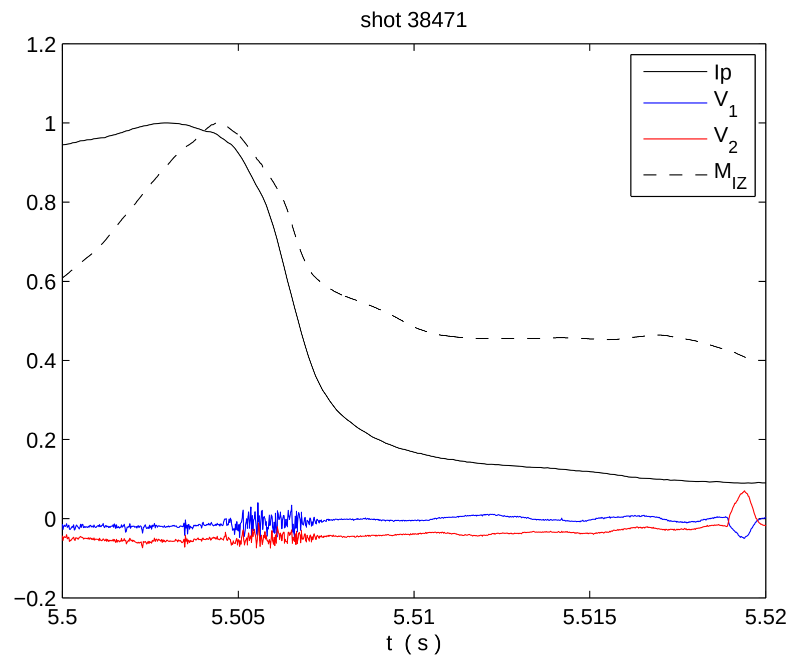
4 types of currents can be distinguished by Xiong tiles.

First measurements of Hiro currents in VDE

Toroidal currents, opposite to the plasma current, predicted by theory (L.Zakharov) and for 2 decades being overlooked in interpretations and simulations of Vertical Disruptions, were measured on EAST in May 2012 (H.Xiong)



Downward VDE



Upward VDE

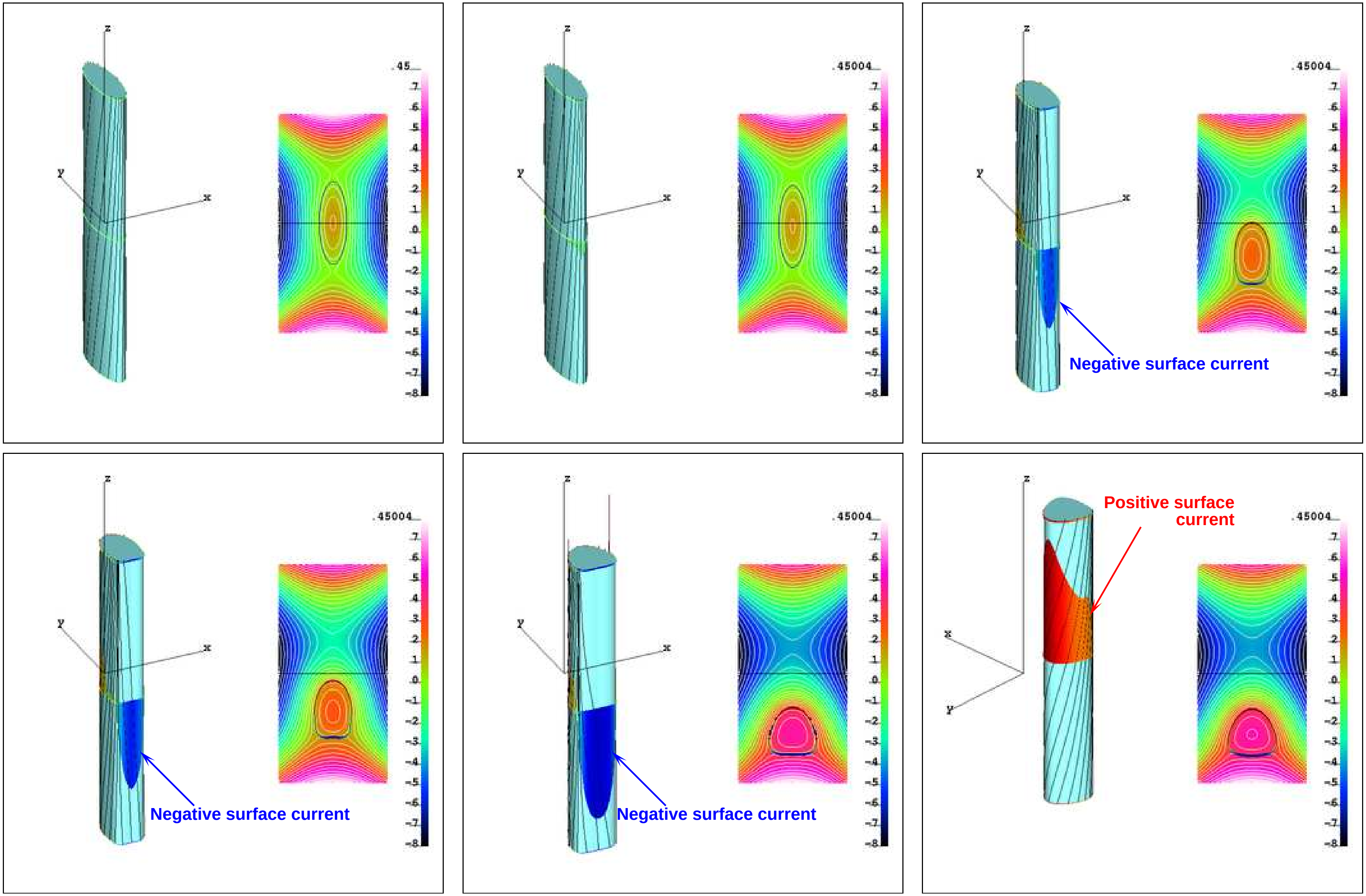
No toroidal asymmetry.

Hiro currents in VDE are NOT SHARED between plasma and the tiles.

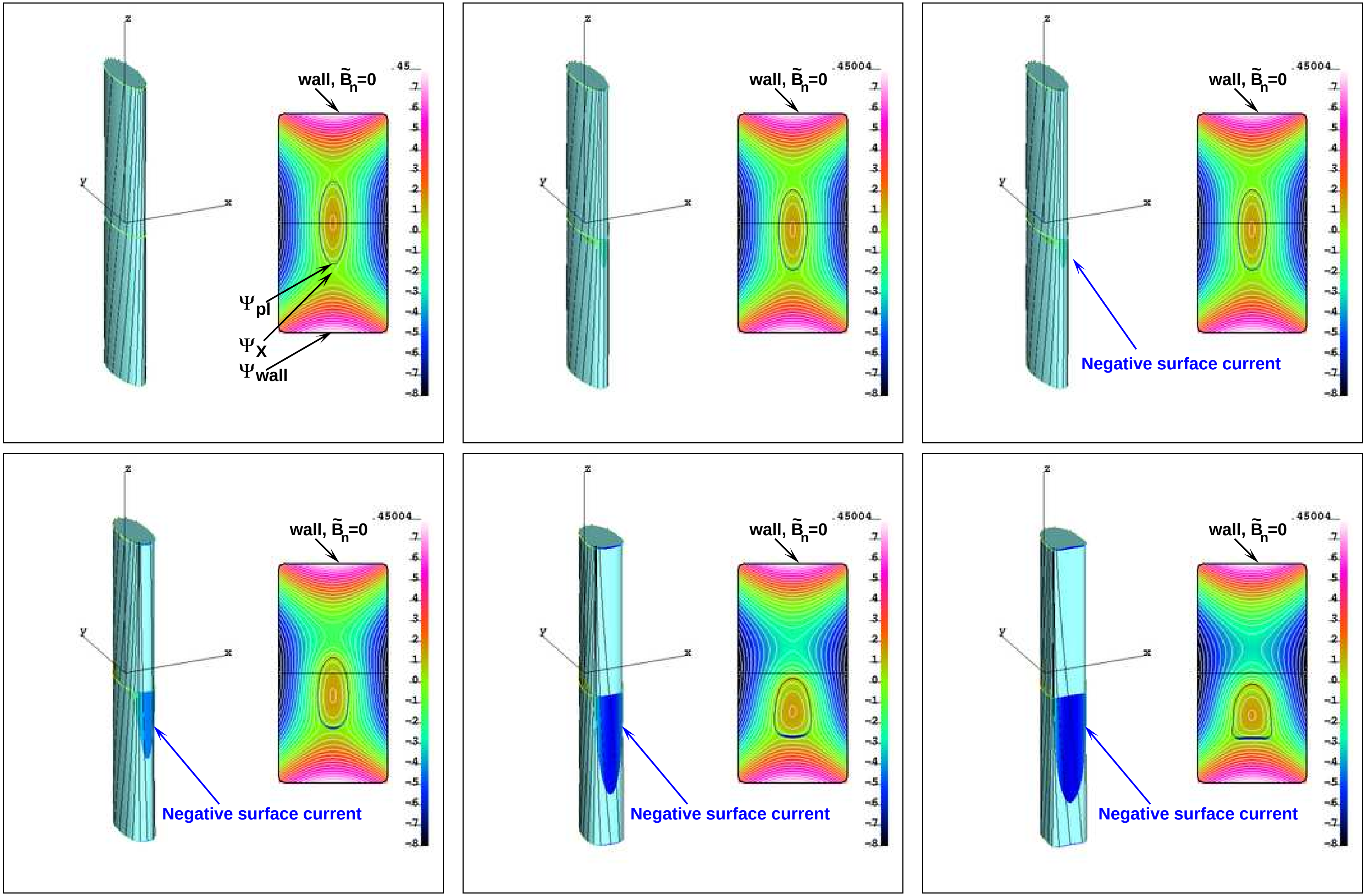
Only certified MHD experts can confused the measured Hiro currents in VDE with the "halo" currents.

Failed first on JET, the fiction of "halo" currents failed now on EAST.

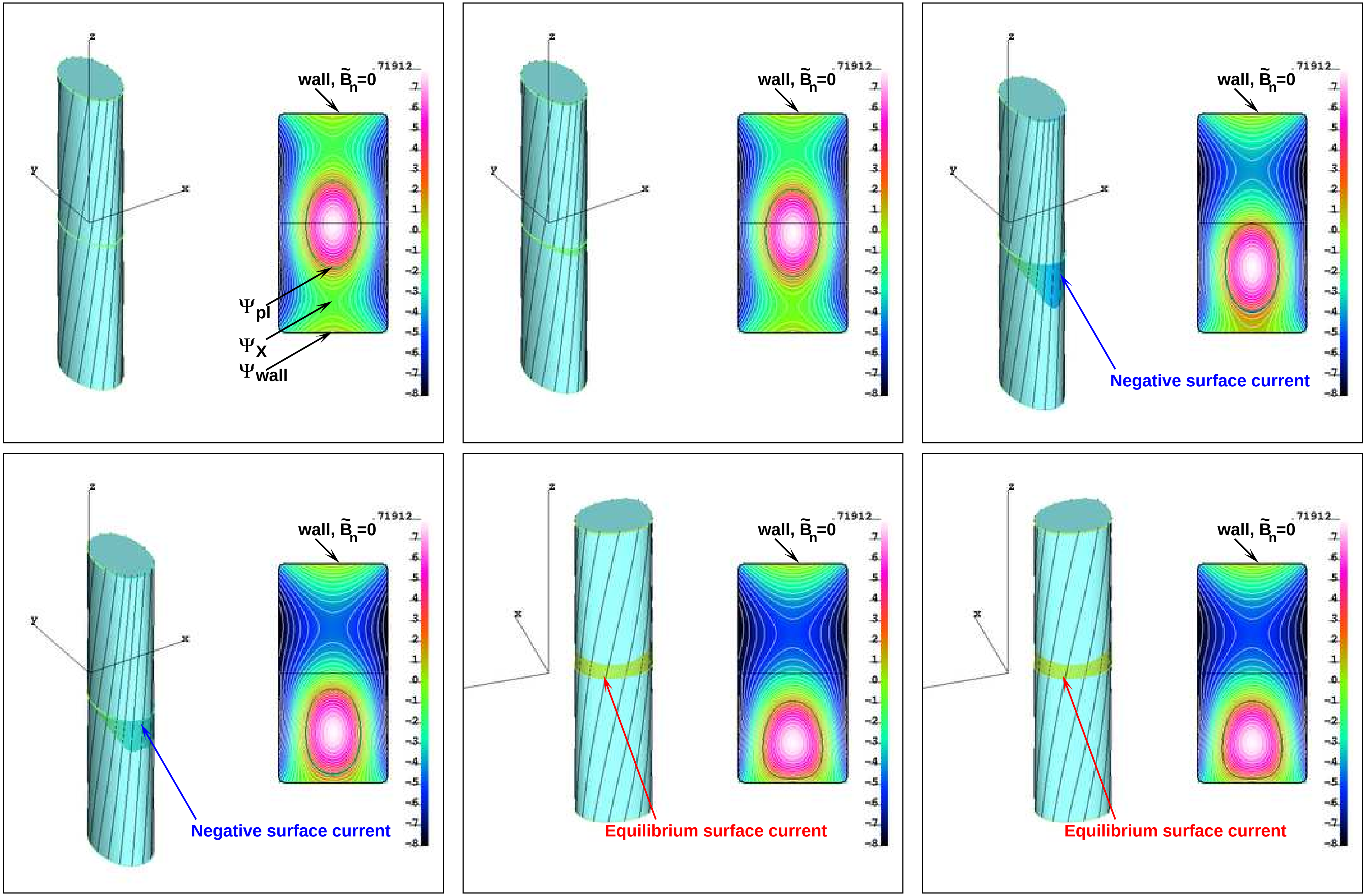
2.5 CylVDE routine for VDE (no wall, no tiles)



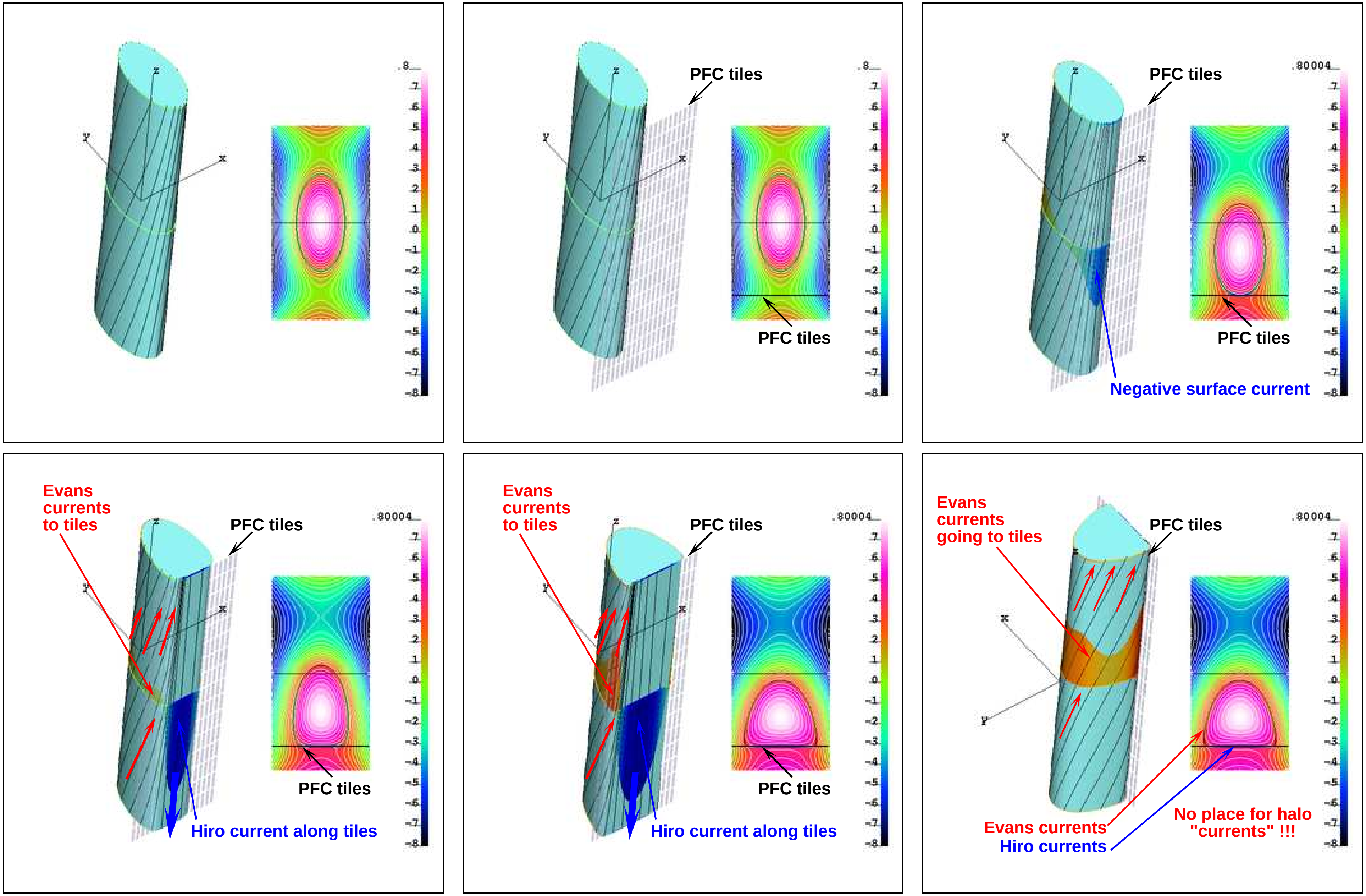
VDE (CylVDE, with a wall, no saturation)



VDE (CylVDE, with a wall, with saturation)

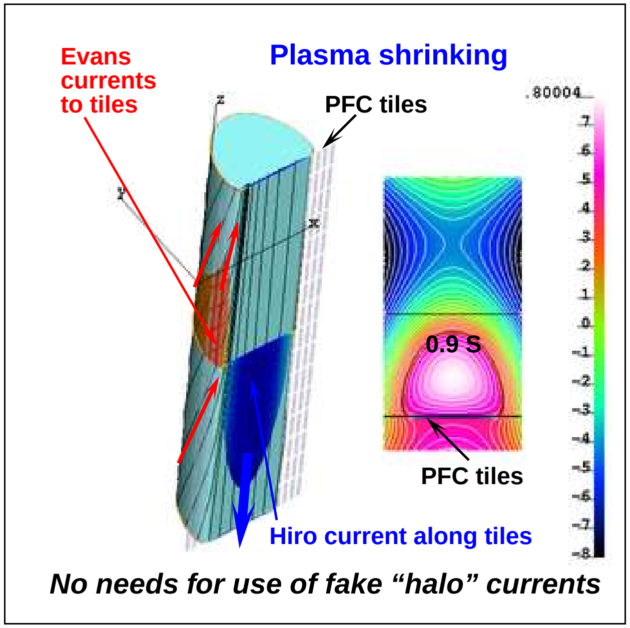


Hiro, Evans currents in VDE (tiles, no wall)

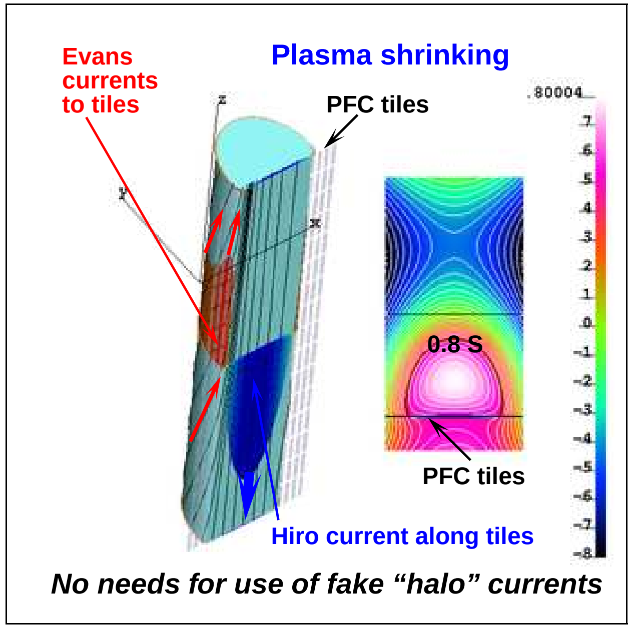


The measured currents to the tiles are Evans currents

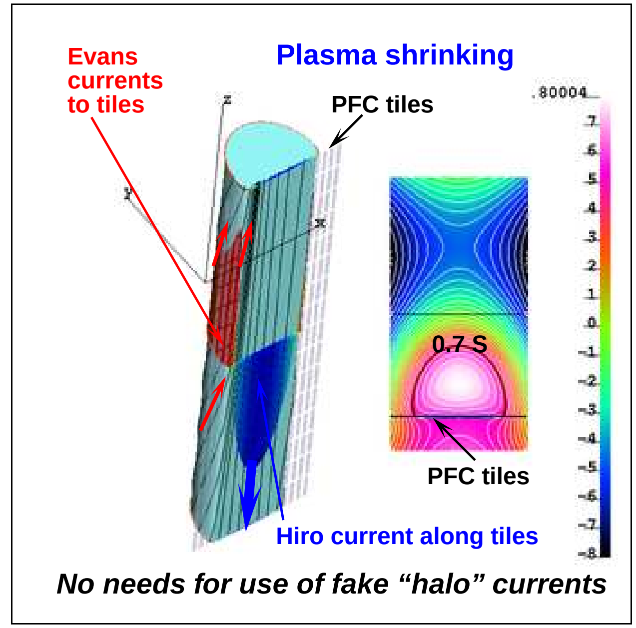
Plasma shrinking due to decay of Hiro currents



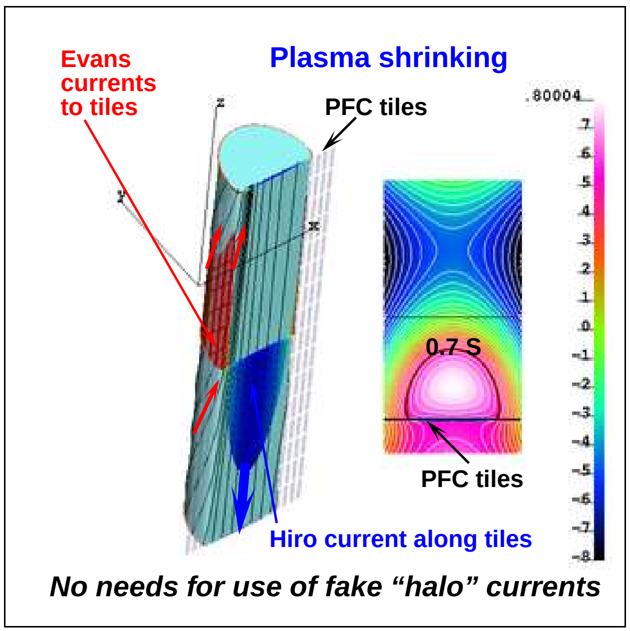
No needs for use of fake "halo" currents



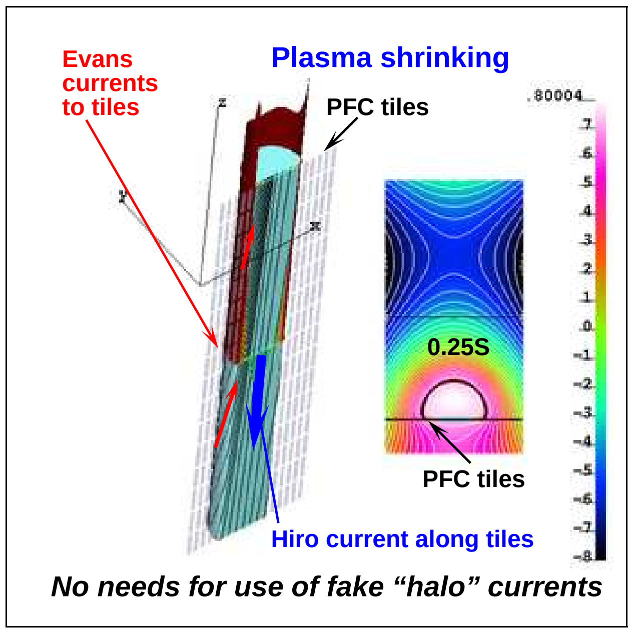
No needs for use of fake "halo" currents



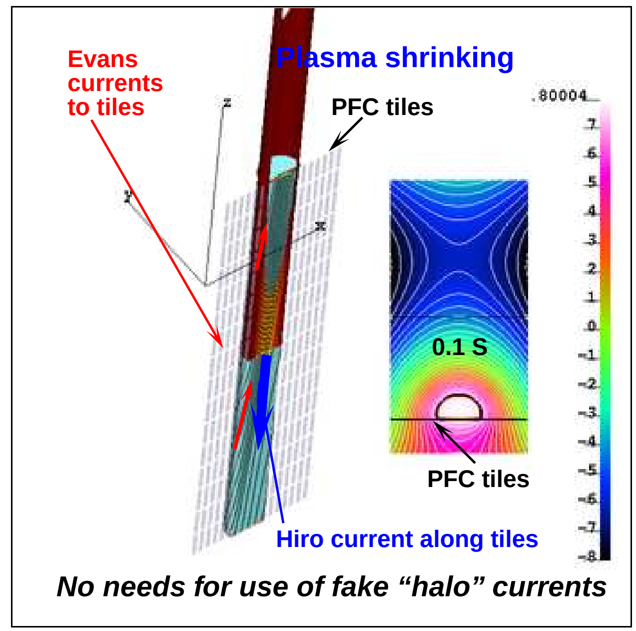
No needs for use of fake "halo" currents



No needs for use of fake "halo" currents



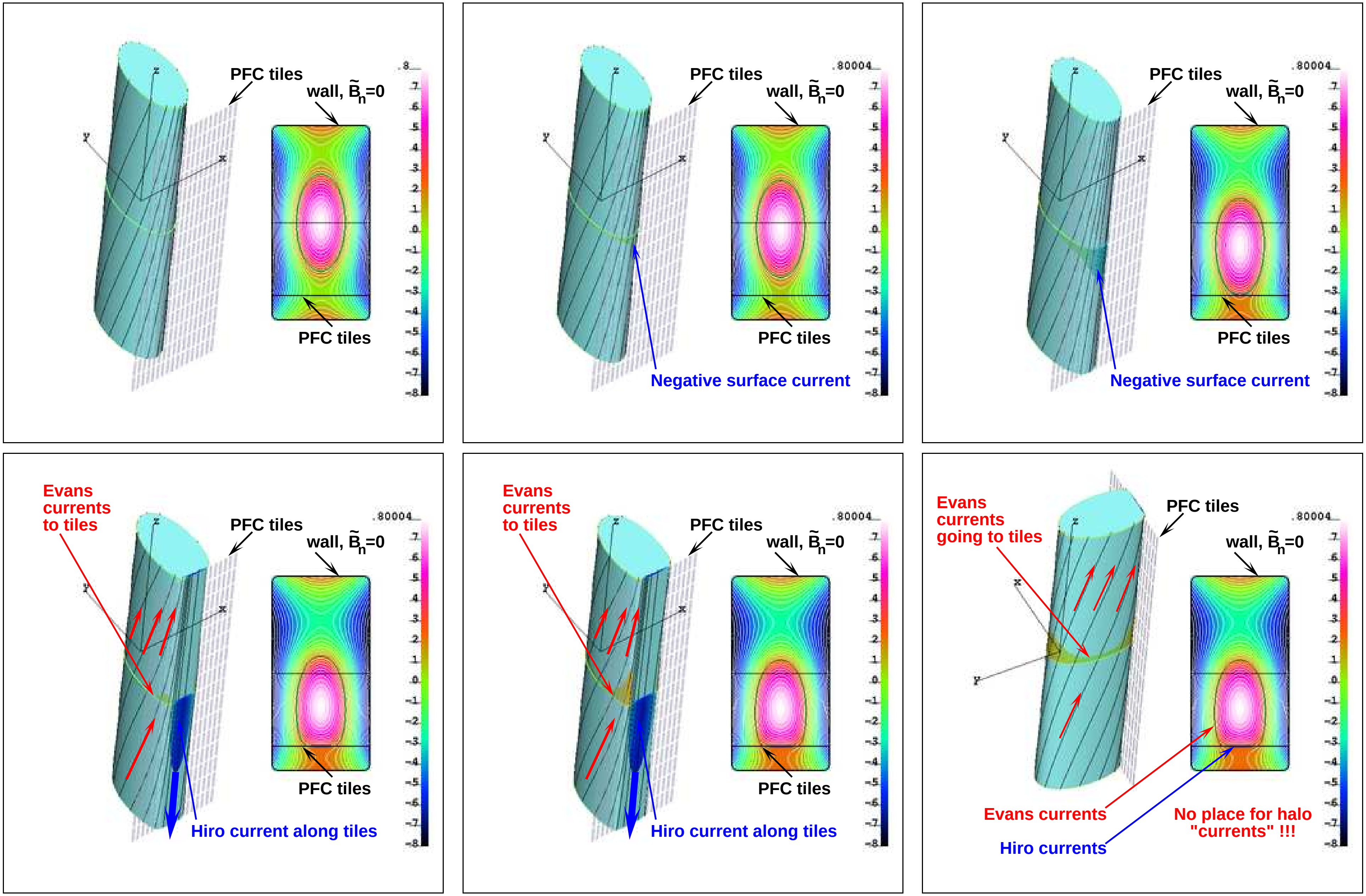
No needs for use of fake "halo" currents



No needs for use of fake "halo" currents

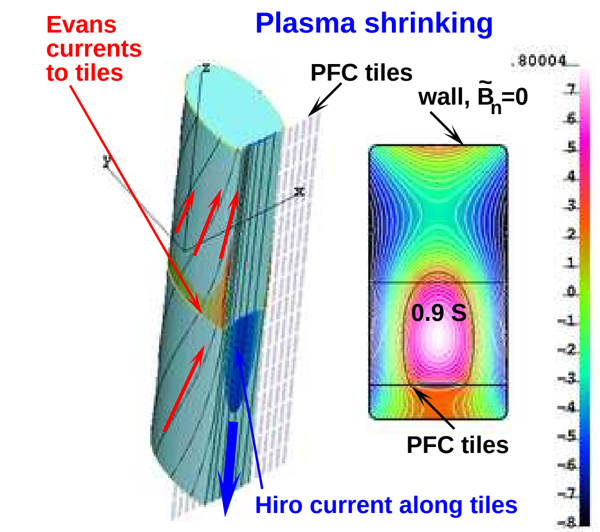
Evans currents explain the tile currents measurements without "halo" current fantasie

Hiro, Evans currents in VDE (tiles and a wall)

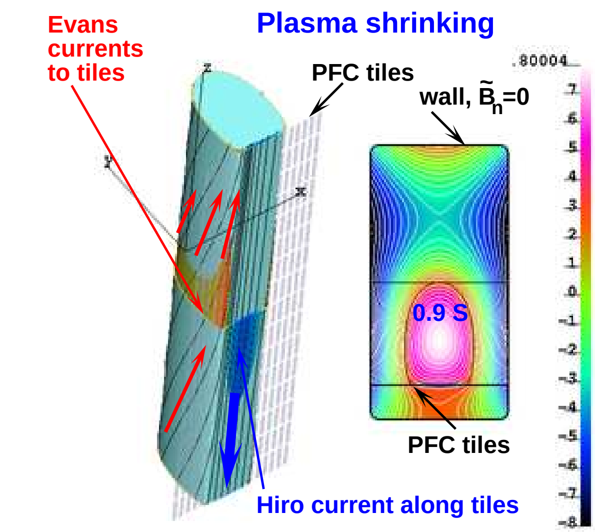


The measured currents to the tiles are Evans currents

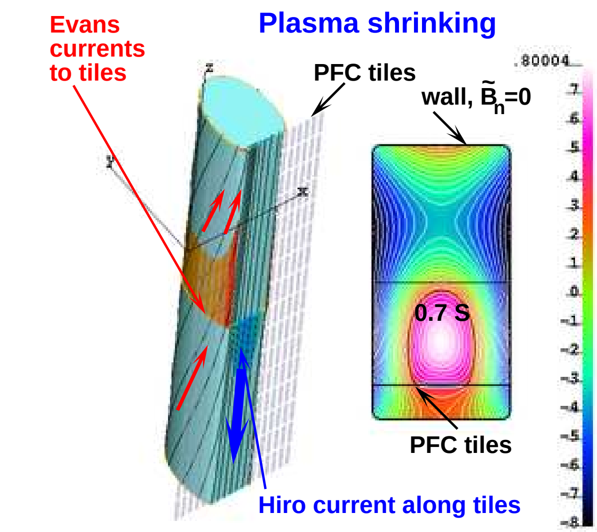
Plasma shrinking due to decay of Hiro currents



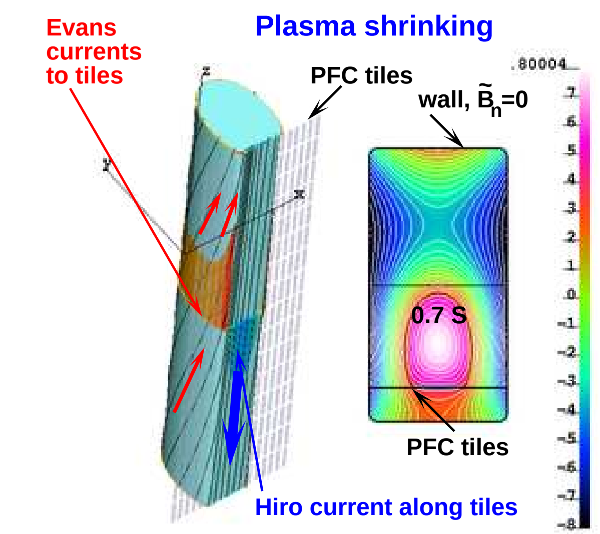
No needs for use of fake "halo" currents



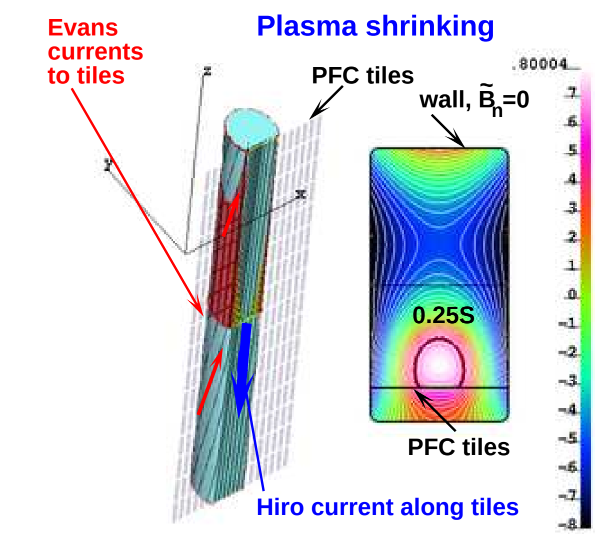
No needs for use of fake "halo" currents



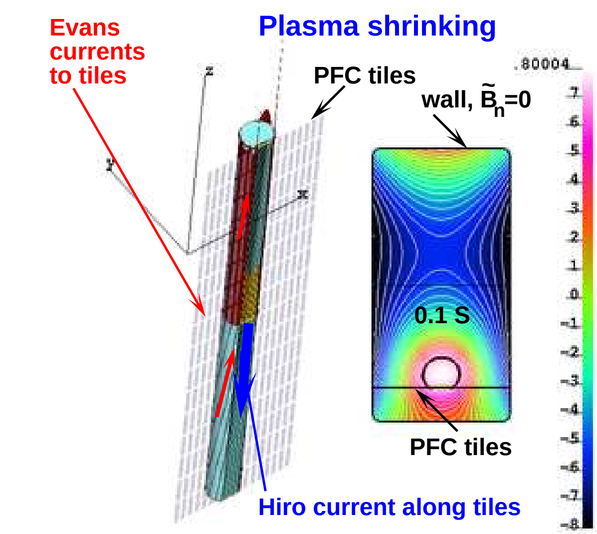
No needs for use of fake "halo" currents



No needs for use of fake "halo" currents



No needs for use of fake "halo" currents



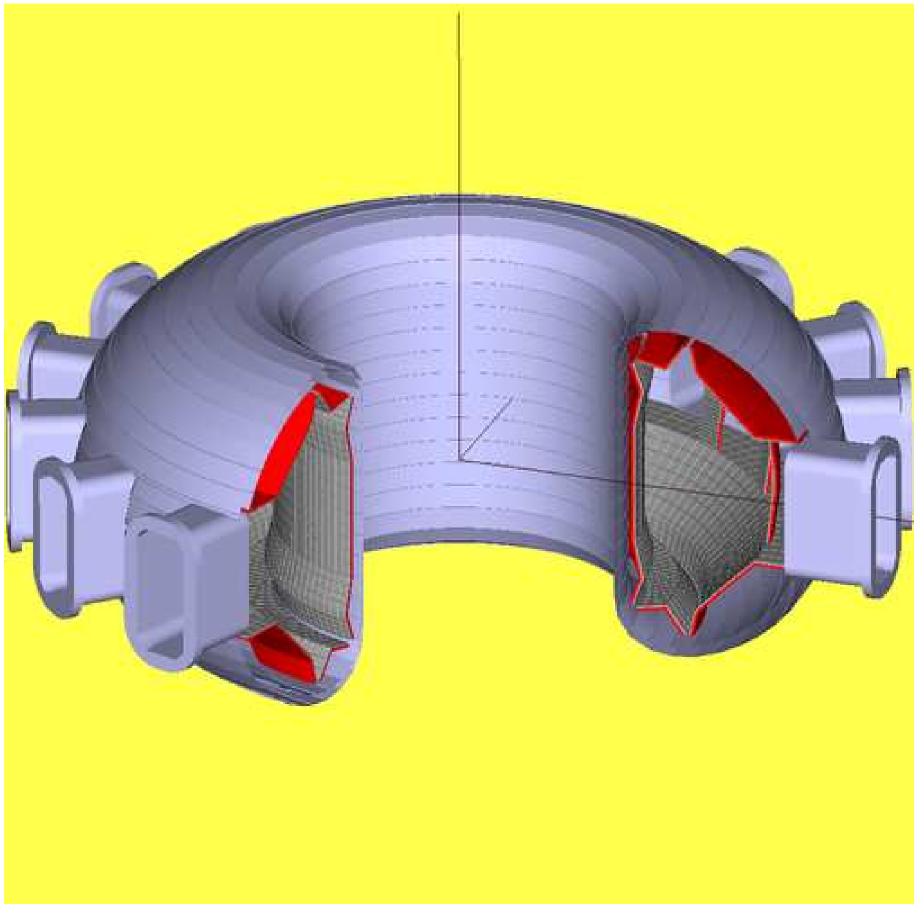
No needs for use of fake "halo" currents

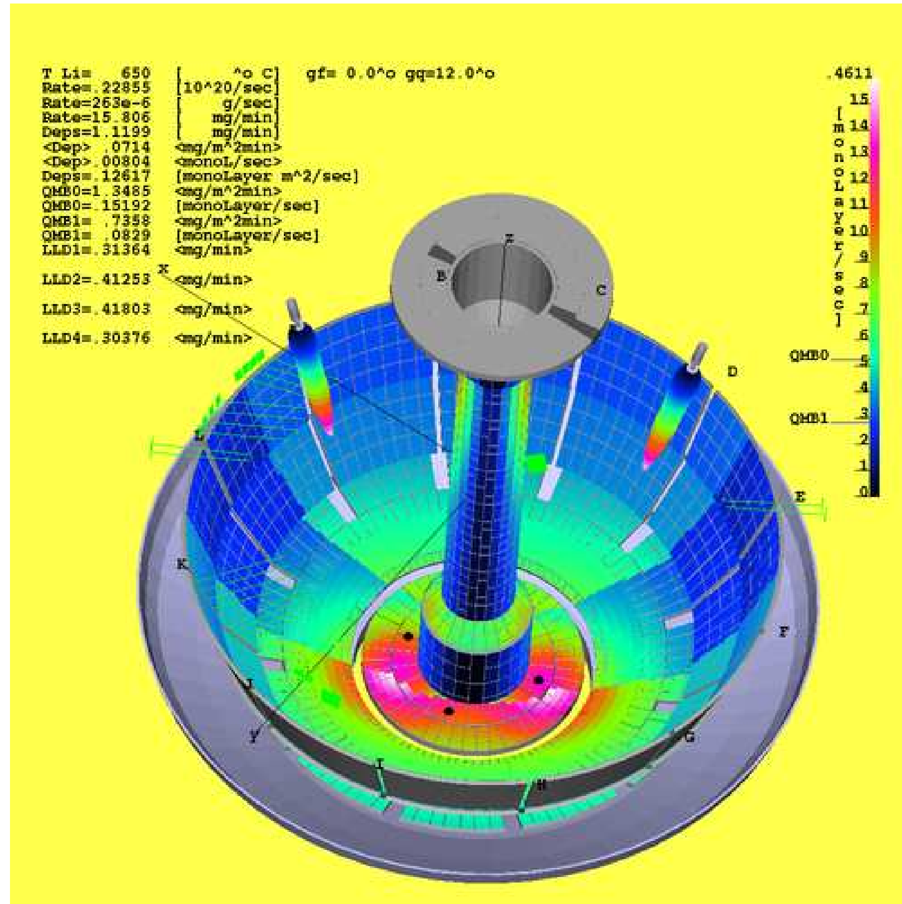
The disruption "experts" of FES are locked into the outstanding misinterpretation of data

2.6 Toward toroidal VDE simulations

Everything necessary for transition to the toroidal VDE simulations in EAST and NSTX is in our possession:

1. ***CylVDE provides the structure and the algorithm;***
2. ***3-D model of EAST and NSTX VV and 3-D Shell current simulation code Cbshl;***

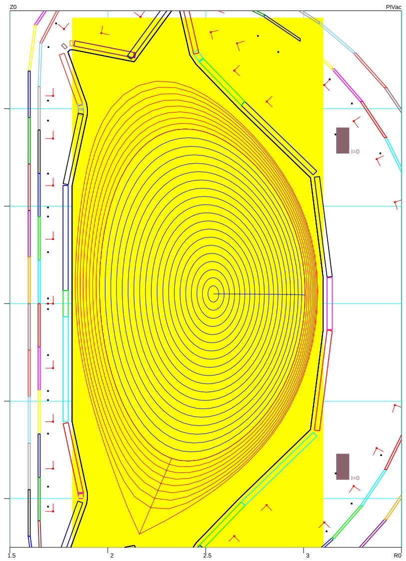




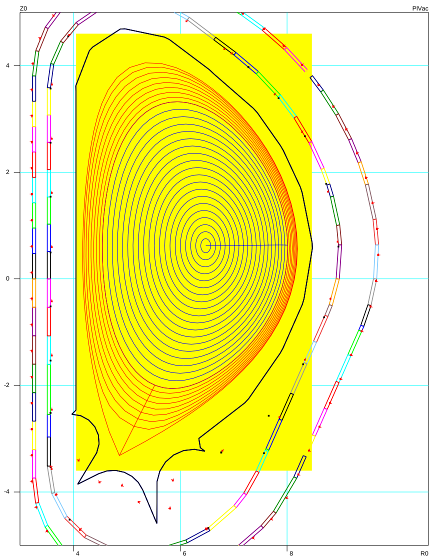
3. ***The best equilibrium code system ESC-EEC as a solver for core equilibrium.***

ESC-EEC code system

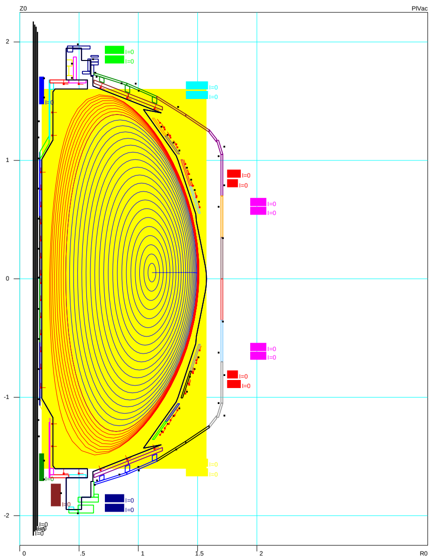
ESC-EEC system, based on flux coordinates, works for any tokamak configuration



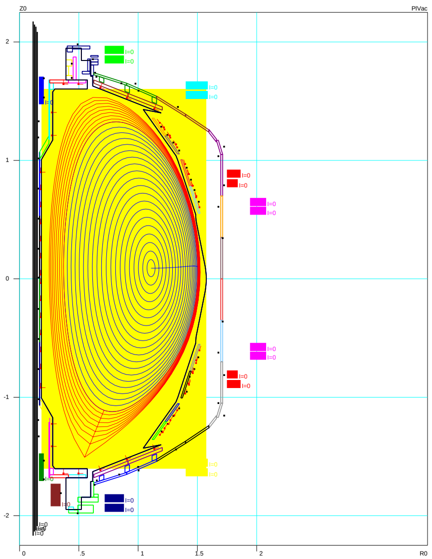
EAST $\beta = 2.5\%$



ITER $\beta = 6\%$



NSTX $\beta = 35\%$



NSTX $\beta = 33\%$

Finite elements at the low field side become much narrower than the plot line. EEC still works !

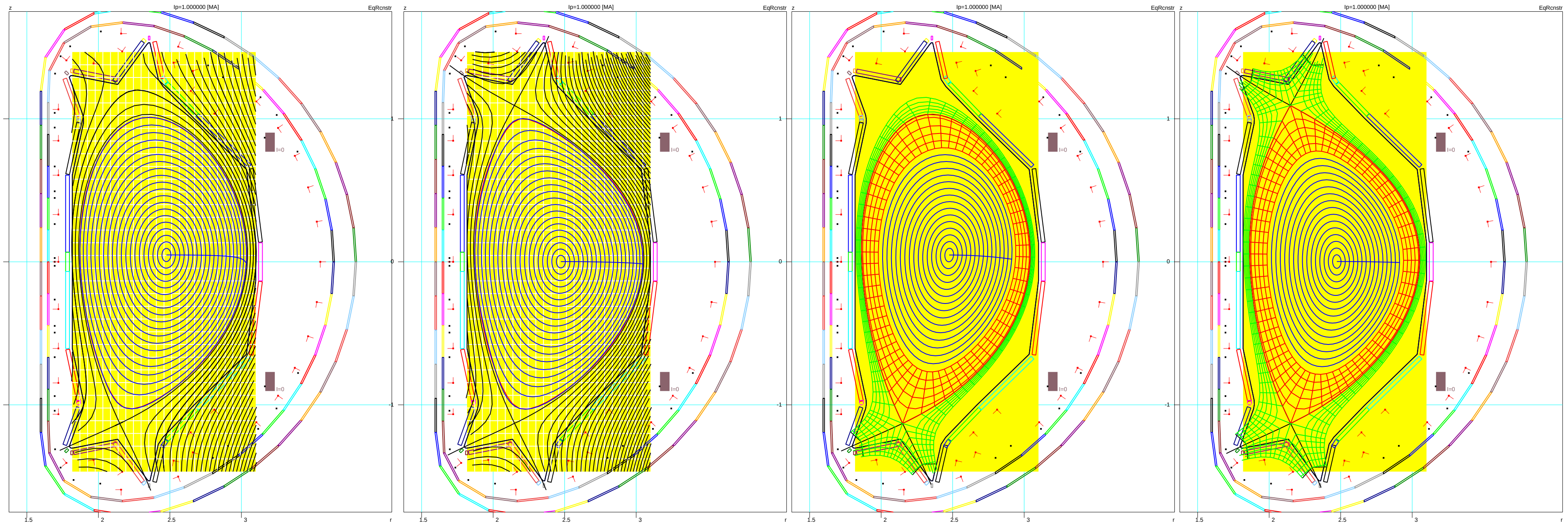
Optimal link of two codes:

- ***Core plasma (blue region) equilibrium is calculated by ESC using Fourier representation***
- ***Edge equilibrium (red region) is calculated by EEC using Hermite elements***
- ***Continuity of magnetic fluxes and fields are provided through a virtual boundary***

Free-boundary equilibria with ESC-EEC

ESC-EEC can calculate free-boundary equilibria in both $r-z$ and flux coordinates

The Equilibrium Spline Interface (ESI) is developed for equilibrium codes instead of present mess in interfacing



(a) ID=00,1,00,00,00 (b) ID=00,01,00,00,00 (c) ID=00,40,00,00,23 (d) ID=00,40,00,00,23

Examples of EAST free boundary equilibrium configurations with (a,c) single and (b,d) double null separatrixes calculated by ESC-EEC.

a),b) Interface IDs for equilibria with $r-z$ coordinate data;

(c),d) ESI IDs for equilibria with the core, edge and vacuum flux coordinate data

In 2011-12, two Theory Dept. reports (one by Boozer's, and another by M.Bell's committees) have been fabricated to prase M3D and TSC as disruption simulation codes. Intentionally biased, both approved the faulty approach of M3D and TSC, while complementing each mentioning of Hiro current theory exclusively by negative comments.

The spirit of reports was expressed by S.Jardin (ITPA-MHD Meeting, Padova, Oct. 4-7, 2011)

> **In 2010, a single scientist in the U.S. fusion community was repeatedly making the following claim (and being quite vocal about it)**
>
> **"...the present numerical codes (M3D, NIMROD) are not applicable of simulating disruptions because of their "salt-water" boundary condition Vnorm = 0, irrelevant to tokamak plasma. For almost 4 years this boundary condition was not corrected. In fact, it represents a fundamental flaw of numerical scheme, making it not suitable for plasma dynamics in tokamaks."**
>
> **This claim was not backed-up by any mathematical, physical, numerical, or experimental analysis, but arose primarily because the code's results did not support that scientist's theory of disruptions.**

Wow, so great ! The problem is that everything in the last paragraph is upside down.

In fact, while comprehensive JET data analysis, physics of Hiro currents, their explicit mathematical expressions and DSC simulations

revealed the GIGO nature of M3D,
the EAST Hiro current measurements
have proved the GIGO nature of 2-D TSC as well

The set of lies

- **“This claim was not backed-up by any mathematical”** ***is a explicit lie. In fact***

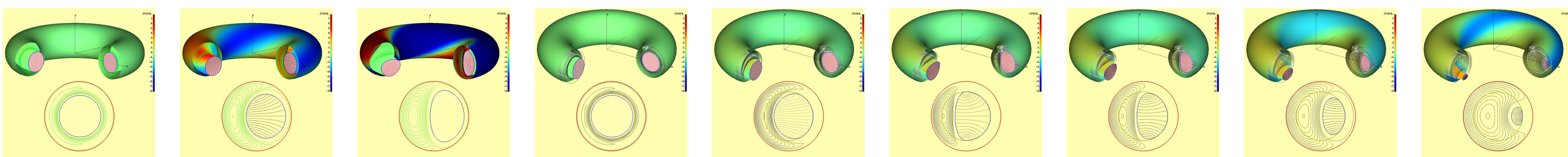
$$\mu_0 \vec{i}_{11} = -2\xi_{11}\frac{B_\varphi}{R}\left(\vec{e}_\varphi + \frac{a}{R}\vec{e}_\omega\right)\cos(\omega - \varphi) \tag{3.1}$$

- **“physical”** ***is an explicit lie. In fact***

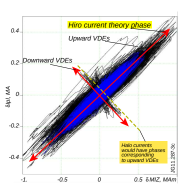
$$-\frac{\partial \vec{A}^{i,surf}}{\partial t} \underbrace{-\frac{\partial \vec{A}^{pl,core}}{\partial t} + VB_\omega \vec{e}_\varphi}_{vanishes\ for\ m=1} - \underbrace{VB_\varphi \vec{e}_\omega}_{driving\ EMF} - \nabla\phi_E^{surf} = \frac{\vec{j}}{\sigma} \tag{3.2}$$

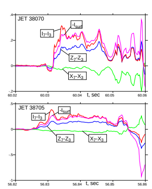
This Faraday law provides the physics of excitation of Hiro currents

- **“numerical”** ***is a explicit lie. In fact***



- **“or experimental analysis”** ***is a explicit lie. In fact***





- **“but arose primarily because the code’s results did not support that scientist’s theory of disruptions.”** ***is lie. In fact, the GIGO codes like M3D and TSC did not affected the development of science of disruptions.***

Since 2007 the boundary condition remains uncorrected for 6.5 years ! In fact, it cannot be corrected.

Two PPPL Theory Dept. reports (2011-2012, 11 authors) have been fabricated in order to hide the failure on multi-M$ M3D, NIMROD, TSC.

1. *"The boundary conditions used in the TSC, DINA, and M3D simulations are appropriate for obtaining an estimate of the maximum of the total force exerted on the wall by the halo current under certain approximations, such as axisymmetry in the TSC and DINA codes, by varying assumed values for the resistance and width of the halo."*
2. *"The assumption in existing simulations that the plasma cannot flow into the wall, $v_n = 0$, is unphysical. . . . the impact of this boundary condition on current simulations is limited to essentially the inertia of the halo plasma, which is NEGLIGIBLE in the overall simulation."*
3. *"Can this boundary conditions be improved to enhance the accuracy of the calculation of the forces ? If so, how ? Yes. Include a model of the electrostatic sheath at the plasmaâwall interface. The complicated geometry of the actual walls should be represented".*

The normal people, unlike the PPPL certified disruption "experts", know the difference between a good and a raptured pipe and will do everything to replace the damaged pipe. Not true for PPPL. It tolerate 6.5 delay with the replacement.

In disruption interpretation the entire fusion community relies on "halo" currents. Our CylVDE simulations ruled them out. The tile currents measured in experiments are the Evans currents.

"Halo" currents are an intrinsic component of M3D and TSC. They cannot reproduce Hiro and Evans currents.

M3D failures in application to ITER

"Wall forces produced during ITER disruptions" by H. R. Strauss, R. Paccagnella and J. Breslau. PHYSICS OF PLASMAS 17, 082505 (2010):

- ***"The initial state is an ITER reference equilibrium, FEAT15MA, written to a file in EQDSK (Ref. 19) format."***
- ***"The wall resistivity for this example had $\gamma\tau_w \simeq 1$ and the current enhancement was $I/I_0 = 1.6$."***

In order to expose the internal $m/n = 1/1$ mode as a candidate for disruption, the authors behind the scene simply enhanced the ITER plasma current from 15 MA to 24 MA !

"Sideways wall force produced during tokamak disruptions" by H. Strauss, R. Paccagnella, J. Breslau, L. Sugiyama and S. Jardin. Nucl. Fusion 53 (2013) 073018

- ***"The VDE expels magnetic flux until the last closed flux surface has safety factor $q \simeq 2$, which destabilizes a fast growing mode with dominant mode number (m, n) = (2, 1). The wall force depends strongly on $\gamma\tau_{wall}$, where γ is the mode growth rate and τ_{wall} is the resistive wall penetration time, and is largest for $\gamma\tau_{wall}$ = constant, which depends on initial conditions.***

In JET VDE there is no minimal indication of generation of an appreciable $m/n = 2/1$ mode. (See Fig. 7, Gerasimov et al, "Plasma current asymmetries during disruptions in JET", Nucl. Fusion, 14)

While M3D failed with 3-D effects in disruptions, TSC failed to recognize both Hiro and Evans currents.

4 Summary. Returning fusion to its original idea

The original idea of magnetic fusion in the 1950s was to use the magnetic field in order to

- ***confine the high temperature plasma and***
- ***insulate it from material walls.***

Tokamaks and Neutral Beam Injection have implemented the first part of the fusion idea

The second part was never implemented: the plasma is in strong contact with the Plasma Facing Components (PFC)

All major problems of magnetic fusion: insufficient confinement and stability, unpredictable plasma-wall interaction, incompatibility with the continuous burning regime, etc - are related to the relatively low plasma edge temperature.

LiWF vs FES

The real power of the second part of the magnetic fusion idea was realized only recently (1998-2009), with the development of the Lithium Wall Fusion (LiWF).

1. ***LiWF resolves the confinement problem:*** ***the best possible, particle diffusion based, confinement regime with expected by order of magnitude better confinement time then presently achieved***

   FES: *misrepresents the confinement as core transport problem, never answered the basic question "Why the is a core and a pedestal ?"*

2. ***LiWF gives*** ***understanding of the plasma edge and the temperature pedestal***

   FES: *trapped to the fake notion of the "edge transport barrier" never understood the plasma edge*

3. ***LIWF:*** ***the best possible stability regime (no sawteeth, ELMs, density limit disruptions), and well predictable plasma profiles***

   FES: *a mess with stability when everything can be unstable. Mess in interpretations based on halo currents. Lack of MHD numerical model for macroscopic plasma dynamics*

4. ***LIWF gives:*** ***only practical way to resolve the power extraction problem***

   FES: *relies on a miracle with material development without touching the faulty plasma physics approach*

5. ***LIWF suggests innovative approaches*** ***heat flux, fueling, stationary burning plasma, etc***

   FES *has no clue how to handle these "unresolvable" problems*

6. ***Finally***

   LiWF gives ***a realistic possibility of 100 MW (R/a=4 m/1 m, Ipl=5 MA) DEMO tokamak facility with Qelectric >1***

   FES *has no clue what what is a magnetic fusion DEMO*

What is the reality ?

The FES program has to be reoriented to the LiWF. It is right and critical time.

After EAST, the near term goal is the implementation of the LiWF regime in the forthcoming DT experiments on JET with the objective of achieving Q$_{DT}$ >5.

We should not miss the chance of demonstrating the real fusion on JET in 3-4 years.

What are the hopes? The following is an example of how OFES is educated by PPPL:

"Conducting safe plasma experiments with lithium is key to understanding its role in control of the plasma edge and wall interaction, and its possible utility for future fusion confinement systems.

At PPPL, we are pursuing lithium and liquid PFCs as an opportunity to provide a robust power and particle-exhaust boundary ***which may also improve confinement****."* ***Wow !***

Obsessed with the idea to revive the corpse of NCSX, PPPL management is totally blind to the fact that all Li experiments dramatically improved not only confinement, but also MHD stability, current drive, predictability of the plasma regime - in accordance with robust theory.

At the same last 15 years of LiWF, the baseless idea of confinement by a single magnetic surface was given a full green light to destroy PPPL experimental base together with PPPL fusion future.

As the result, LiWF is funded exclusively by LDRD money;

- ***After installation of FLiLI limiter on HT-7 and experiments, our LDRD funding was interrupted for half year and eliminated the development of the freeze valves for EAST;***
- ***This February after delivery of $^{7}_{24}$FLiLi limiter to ASIPP - another half year interruption, eliminated the development of wetting technique, thus, arranging a failure of our the first $^{7}_{24}$FLiLi experiment***

The future of fusion depends on the success of wetting on EAST, rather than on Theory activity. If the chance is missed, then JET will fail with even $Q_{DT} = 1$. It is easy to predict that magnetic fusion, as we know it, will be dead.

In turn, we are eliminating the non-sense of "halo" currents - the backbone of the current plasma stability concept